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Update of Part 61 Impacts Analysis Methodology

Methodology Report

ORIGINAL

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Envirosphere Company

**Prepared for
U.S. Nuclear Regulatory
Commission**

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Vol. 1

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NOTICE TO: Readers of NUREG/CR-4370 - Update of Part 61 Impacts Analysis Methodology

This report presents an updated version of the analysis methodology used by NRC in the draft and final environmental impact statements on the regulation 10 CFR Part 61 ("Licensing Requirements for Land Disposal of Radioactive Waste"). The updated methodology was completed prior to passage of the Low-Level Radioactive Waste Policy Amendments Act of 1985, which assigns responsibility for disposal of waste that exceeds Class C concentrations to the Department of Energy (DOE), and requires DOE to develop recommendations for implementing this new responsibility. Further NRC actions to develop the regulatory framework for such waste will necessarily take DOE's recommendations into account.

The updated analysis methodology is implemented using a number of computer codes written for use on the IBM PC. Waste management costs and radiological impacts are calculated based on a user-specified combination of disposal options, including those for waste processing and packaging, disposal methods, and disposal site environments.

The updated analysis methodology is believed to be most useful as a tool to rapidly analyze and compare disposal options on a generic basis. An example use could involve analyzing alternatives for disposal of low-level waste that exceeds Class C concentrations. Another example use could involve case-by-case sensitivity studies of the general levels of impacts that would be associated with disposal of particular individual waste streams by different methods.

However, caution is advised in using the analysis methodology in a site-specific application, where site-specific models, radionuclide inventories, disposal methods, and environmental parameters would have to be considered. Caution is also advised in interpreting the absolute magnitudes of the calculated results. The magnitude of an individual set of results is less meaningful than the comparison of one set of results (for a given set of disposal options) with another set of results.

NRC is also interested in continued improvement of its calculational ability. Therefore, any information that would improve the analysis methodology data base and calculational algorithms would be appreciated. NRC is especially interested in additional information on the characteristics of individual low-level waste streams generated by commercial licensees.

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ABSTRACT

Under contract to the U. S. Nuclear Regulatory Commission, the Envirosphere Company has expanded and updated the impacts analysis methodology used during the development of the 10 CFR Part 61 rule to allow improved consideration of the costs and impacts of treatment and disposal of low-level waste that is close to or exceeds Class C concentrations. The original impacts analysis methodology is described in the report "Data Base for Radioactive Waste Management" (NUREG/CR-1759), which was prepared under contract to NRC by Dames and Moore.

The modifications described in this report principally include: (1) an update of the low-level radioactive waste source term, (2) consideration of additional alternative disposal technologies, (3) expansion of the methodology used to calculate disposal costs, (4) consideration of an additional exposure pathway involving direct human contact with disposed waste due to a hypothetical drilling scenario, and (5) use of updated health physics analysis procedures (ICRP-30).

Volume 1 of this report describes the calculational algorithms of the updated analysis methodology, while Volume 2 describes the computer codes written to implement the updated analysis methodology plus provides some example problems. The computer codes are written for operation on an IBM personal computer, and are available from the Radiation Shielding Information Center at Oak Ridge National Laboratory.

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1.0 INTRODUCTION

The purpose of this report is to present an updated version of the analysis methodology (Ref. 1) used by the Nuclear Regulatory Commission (NRC) to analyze alternatives in the draft and final environmental impact statements (EIS) (Refs. 2, 3) on the regulation 10 CFR Part 61 ("Licensing Requirements for Land Disposal of Radioactive Waste") (Ref. 4). Compared to the original Part 61 analysis methodology, this updated version:

- Considers the radiological, physical, and chemical characteristics of low-level waste in more precise detail, giving special emphasis to wastes that are close to or exceed Class C concentrations as defined in 10 CFR Part 61;
- Considers a number of potential near-surface disposal methods that might be suitable for such waste; these disposal methods include trench disposal plus a number of alternatives such as concrete structures;
- Updates where applicable the scenarios and models used to calculate radionuclide transport through the environment and to determine potential radionuclide impacts when contacted, inhaled, or ingested by humans; and
- Updates where applicable the models used to determine economic and other impacts on licensees.

This updated analysis methodology may be potentially used in two different ways:

- As a source document in a rulemaking effort on the definition of high-level waste by providing a systematic means of analyzing alternatives; and
- As a methodology that can be used to generically analyze disposal of individual wastes that exceed Class C concentrations on a case-by-case basis.

An introduction to the updated Part 61 analysis methodology is presented in this Chapter 1.0. Sections 1.1 and 1.2 respectively present a background and discussion which place the concepts introduced in the above paragraphs into an organized framework. Section 1.3 addresses the general approach to the updated analysis methodology while Section 1.4 presents a brief summary of the report contents.

1.1 Background

The background to this study is dominated by the past definitions of low- and high-level waste, and the recent initiatives on their regulation.

Regulation of Low-Level Waste

Resolution of the technical, institutional, social, and political issues surrounding the disposal of low-level radioactive waste has been called by many an important national goal. One of the milestones on the way to meeting this goal was the passage of the Low-Level Radioactive Waste Policy Act in 1980 (Pub. L. 96-573). This Act establishes the principle that disposal of low-level waste is a state responsibility while reconfirming the principle that disposal of high-level waste is a federal responsibility. The Act provides that states may establish regional compacts for the purpose of low-level waste disposal and, once a compact has been approved by Congress, exclude waste from outside the borders of the compact. This Act also defines low-level waste as radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in section 11 e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste).

Another significant milestone in achieving the above goal was the promulgation by the NRC of the regulation 10 CFR Part 61 ("Licensing Requirements for Land Disposal of Radioactive Waste") on December 27, 1982 (Ref. 4). At the time the NRC regulatory development process was initiated, there was a well documented need for comprehensive national standards and technical criteria for the disposal of low-level waste. The absence of sufficient technical criteria and standards was seen to be a major deterrent to the siting of new disposal facilities by states and compacts.

The Part 61 regulation was supported by the draft and final environmental impact statements (EIS) (Refs. 2, 3) prepared by the NRC staff; several alternatives for rulemaking were considered in these EIS's, analyzed numerically in terms of economic costs and radiological impacts, and compared. An analytical methodology was created to perform these rulemaking analyses and the initial version of this analysis methodology (used for the draft Part 61 EIS) is described in reference 1. Considerable modifications to the initial Part 61 analysis methodology were made to perform the calculations for the final EIS.

The Part 61 regulation establishes procedural requirements, institutional and financial requirements, and overall performance objectives for land disposal of radioactive waste, where land disposal may include a number of possible disposal methods such as mined cavities, augered holes, engineered bunkers, or shallow land burial. This regulation also contains technical criteria (on site suitability, design, operation, closure, and waste form) which are applicable to near-surface disposal, which is a subset of the broader range of land disposal methods. Near-surface disposal is defined as disposal in or within the upper 30 meters of the earth's surface, and may include a range of possible techniques such as concrete bunkers or shallow land burial. The rule is intended to be a non-prescriptive regulation with the result that the Part 61 technical criteria are written in relatively general terms, leaving a need for additional work in interpreting these criteria for various specific near-surface disposal methods. The regulation defines low-level waste in the same manner as the Low-Level Radioactive Waste Policy Act.

A waste classification system was also instituted in the regulation which establishes three classes of waste suitable for near-surface disposal: Class A, Class B, and Class C. Limiting concentrations for particular radionuclides were established for each waste class, with the highest class being Class C. The Class C limits were established based on NRC's understanding, at the time of the rulemaking, of low-level waste characteristics and potential disposal methods, and are applicable to all potential near-surface disposal methods. However, the numerical calculations performed to establish the limits were principally based on postulated use of one near-surface disposal method: shallow land burial. The Class C limits are therefore conservative since there may be a number of near-surface methods that have greater confinement capability (and higher disposal costs) than shallow land burial.

The regulation states that waste exceeding Class C concentrations is considered to be "not generally acceptable for near-surface disposal," where this is defined in paragraph 61.55(a) as "waste for which waste form and disposal methods must be different, and in general more stringent, than those specified for Class C waste" (Ref. 4). Thus, waste exceeding Part 61 Class C concentrations has been generally excluded from near-surface disposal and is being held in storage by licensees. (This amounts to less than 1% of the approximately 3,000,000 ft^3 of commercial low-level waste annually being generated.) Given the current absence of specific requirements for disposal of waste exceeding Class C concentrations, the regulation allows for evaluation of specific proposals for disposal of such waste on a case-by-case basis. The general criteria to be used in evaluating specific proposals are the Part 61 performance objectives contained in Subpart C of the regulation.

Regulation of High-Level Waste

By authority of the Energy Reorganization Act of 1974 (Pub. L. 93-438), the NRC exercises licensing authority over the following waste facilities:

- Facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under the Atomic Energy Act; and
- Retrievable Surface Storage Facilities (RSSF) and other facilities authorized for the express purpose of subsequent long-term storage of high-level radioactive waste generated by the Department of Energy (DOE), which are not used for, nor are part of, research and development activities.

Based on this authority, the NRC developed and adopted regulations that govern the licensing of DOE activities at geologic repositories for disposal of high-level waste. These regulations are codified in 10 CFR Part 60 (Ref. 5), and define high-level waste in section 60.2 of this rule as follows:

> "High-level radioactive waste, or HLW, means: (1) irradiated reactor fuel, (2) liquid wastes resulting from the operation of

> the first cycle solvent extraction system, or equivalent, and the concentrated waste from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and (3) solids into which such liquid wastes have been converted."

More recently, the Nuclear Waste Policy Act of 1982 (Pub. L. 97-425) provides for the development of repositories for the disposal of high-level radioactive waste and establishes a program of research, development, and demonstration. This law defines the term "high-level radioactive waste" as:

> The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and
>
> Other highly radioactive material that the Commission (NRC), consistent with existing law, determines by rule requires permanent isolation.

1.2 Considerations

As indicated in the previous section, the legal and regulatory definitions of both high-level and low-level wastes are given in an imprecise manner. Both definitions address the source of the waste rather than the specific radioactive contents. This has led to some problems.

One problem is that while the above source oriented definitions were developed with an understanding of the relative difference in hazard potential presented by the two types of waste, there can be some overlap. That is, both high-level and low-level wastes may contain a wide range of radionuclides and radionuclide concentrations. Occasionally these radionuclide concentrations overlap, so that some wastes defined as low-level wastes may have radionuclides in concentrations that exceed those in some high-level wastes. A definition of both types of waste is therefore needed which is more precisely based on radioactive hazard.

Based on past experience with low-level waste disposal and 10 CFR Part 61, it would appear that a limit based on the total activity or concentration of particular radionuclides of concern would be most suitable. That is, concentration oriented definitions of high- and low-level wastes would appear to be more suitable for the protection of public health and safety since, in many cases, public exposures are linearly dependent on radionuclide concentrations.

A second problem is the existence of "orphan wastes" -- i.e, low-level wastes generated in the commercial sector for which disposal criteria are undefined. Prior to the promulgation of the Part 61 regulation, there was no uniform waste classification system, although all operating low-level waste disposal sites had adopted license conditions that prohibited disposal of waste containing transuranic material in concentrations exceeding 10 nCi/g. Waste exceeding 10 nCi/g was "orphaned" since: (1) there was no commercial (licensed by NRC or an Agreement State) capacity for low-level waste disposal, and (2) the Department of Energy took the position that as long as it was not "high-level waste" the Department had no legal authority to accept the waste for storage and eventual disposal. Licensees therefore had no place to dispose this waste.

Resolution of this Catch-22 situation was not accomplished with the promulgation of 10 CFR Part 61. Wastes exceeding Class C concentrations are still defined as low-level waste and are therefore outside of federal responsibility. Licensees holding wastes exceeding Class C concentrations are therefore in the same essential predicament as before. The existence of 10 CFR Part 61, however, does provide a regulatory framework to help resolve the question.

A third problem is a perception by some that the above uncertainties associated with the definitions of high- and low-level wastes present a hinderence to the state compacting process. That is, states may be reluctant to proceed with the compacting process unless the wastes for which the states are responsible for under the Low-Level Radioactive Waste Policy Act are precisely defined.

Resolution of the above problems is the intent of a rulemaking program being initiated by the NRC. (The NRC staff is preparing an Advance Notice of Proposed Rulemaking (ANPRM) which will be published in the Federal Register in late 1985.) Under this rulemaking, the NRC plans to rigorously define a lower limit on high-level waste, and as a natural result, also rigorously define an upper limit on low-level waste. Thus, those wastes which are subject to federal jurisdiction for disposal will be defined based on the hazard potential of the waste, as well as wastes which under existing law are subject to state jurisdiction for disposal.

As discussed in the ANPRM, there may be a number of possible approaches to accomplishing this, some of which would require Congressional action. One approach is to examine low-level wastes that exceed Class C concentrations, as well as various alternatives for low-level waste disposal, to determine whether there exists concentrations of wastes that might be safely disposed by these alternative methods. This could result in a new classification of low-level waste -- "Class D waste" -- which would mark the upper limit of low-level waste disposal. Wastes exceeding Class D limits would be defined as high-level waste. Responsibility for disposal of Class D waste could be that of either the State or Federal government, although the latter approach would require modification of existing law.

In any case, the Part 61 analysis methodology needs to be updated to more precisely define both wastes exceeding Class C concentrations and possible disposal methods.

1.3 General Approach

The general approach taken in the updated Part 61 analysis methodology is very similar to that taken for the original Part 61 analysis methodology as described in reference 1 and heavily modified for use in the final EIS on Part 61 (Ref. 3). The calculational methodology is structured as a number of computer codes, and the selection of a particular code depends upon the type of information desired.

In developing the revised analysis methoodology, two of the principal considerations were the great variety of possible methods for treatment and disposal of low-level waste, as well as the great variety of possible site environments in which this treatment and/or disposal can take place. Another consideration was the need to address the significant pathways for human exposure, and to incorporate any significant and appropriate advances in health physics calculational techniques since the development of the original Part 61 analysis methodology. A very important consideration was the need to consider the depth to which site-specific considerations would enter the calculations. A generic type of analysis implies a less detailed set of input and calculational requirements than a site-specific type of analysis, where the greater availability of real site-specific data would justify a more complete analysis. It was furthermore recognized that the main use of the revised methodology would be for regulatory analyses, in which case the comparative differences between two alternatives are more important that the actual calculated numbers. It was finally recognized that to maximize the usefulness of the calculational methodology, there was a need to maintain a high degree of flexibility in the methodology.

As a result, a two-pronged approach was taken in the updated methodology. First, the methodology makes heavy use of a reference radioactive waste source term and processing techniques, a reference set of site environments, and a reference set of alternative disposal technologies. Second, the computer codes are constructed so that the user can readily modify or add to the reference waste source term, treatment and disposal site environments, and disposal technologies.

The radioactive waste source term consists of approximately 150 individual waste streams which are characterized on a volume, physical, chemical, and individual radionuclide basis. Volumes for these waste streams are considered as a function of four regions comprising the contiguous United States. Reference waste processing operations are also considered. Within each region a reference site is assumed which has environmental characteristics representative of the region. Finally, a number of alternative near-surface disposal technologies and operating practices are assumed.

The analysis methodology developed and described in this report includes procedures to calculate:

- o The occupational exposures and the exposures to members of the public (individuals and populations) resulting from the disposal of low-level waste;

- The occupational and population exposures resulting from the processing of the waste by the waste generator or at a centralized location within a region (assumed to be located at the disposal site), and the transportation of the waste from the waste generators to the disposal site;

- The costs associated with processing, transportation, and disposal of low-level waste; and

- The land area committed to low-level waste disposal.

1.4 Report Contents

This report is presented in two volumes. Volume 1 presents the calculational methodology and the background and assumptions that go into its development. Volume 2 presents the listings for the computer codes, a discussion of their use, and example problems. Volume 1 contents are discussed below.

Main Body

There are five chapters in addition to this introductory Chapter 1.0. Chapter 2.0 is in many ways another introductory chapter, but is very important to understanding the calculational methodology.

Section 2.1 includes a more detailed introduction to the report. It briefly discusses the original Part 61 analysis methodology and its limitations, and also discusses the revised Part 61 methodology using the original as a basis. The section also discusses the components of the data base used in the calculations. These include the radiological source term (the "waste streams"), the waste processing and transportation options considered, the environmental characteristics of the treatment/disposal locations, and the reference disposal technologies. It finally discusses the residual limitations of the revised methodology.

Section 2.2 is a very important section in that it presents the methodology used by the computer codes to signify different treatment and disposal operations -- i.e., the decision indices. The decision indices are integer indices that refer to a specific procedure used in the calculation of impacts or determine a specific value of the environmental parameters. Section 2.3 presents the general approach to the pathway analysis of radiocontaminant release and transfer, and subsequent human exposures. Section 2.4 presents an outline of the general structure of the computer codes.

Chapter 3.0 presents the methodologies used for calculating radiological and economic impacts associated with waste processing (Section 3.1) and transport (Section 3.2) to a disposal site. Waste processing may be carried out either by the waste generator or at a large treatment facility located at the disposal facility site. Radiological impacts are estimated for workers as well as for populations surrounding the processing location if incineration is performed. All waste transportation is assumed to be by truck, and radiological impacts are estimated for transport workers as well as to populations along the transport route. All of the radiological and economic impacts calculated in this chapter are suitable for comparative purposes but should not be construed to represent real impacts to real people.

Chapter 4.0 is the largest chapter in the report and presents the methodology for calculating radiological impacts associated with waste disposal. This includes operational impacts as well as post-disposal impacts.

Section 4.1 is an overview of the chapter, while Section 4.2 presents inadvertent intrusion, Section 4.3 presents groundwater migration, and Section 4.4 presents leachate accumulation ("bathtub") radiological impact scenarios. Section 4.5 presents radiological impacts associated with exposed waste scenarios -- that is, radiological impacts to populations due to hypothetical events such as inadvertent intrusion or erosion which bring waste to the earth's surface where it may be dispersed by wind or water. Finally, Section 4.6 discusses operational exposure scenarios which include routine operational impacts plus accident scenarios.

Chapter 5.0 reviews some of the considerations inherent in classifying waste for disposal, and discusses the applicability of these considerations to the potential establishment of a new waste class (Class D). This chapter also integrates calculation of impacts from the scenarios addressed in Chapter 4.0 with regulatory requirements that could be implemented on waste disposal (e.g., minimum disposal depth, area concentration limits).

Finally, Chapter 6.0 discusses other impact scenarios: disposal costs and land use. Unit costs are presented for five different periods in the life of a disposal facility, and procedures are specified for calculating the land areas involved.

Appendices

Volume 1 of the report includes five appendices. Appendix A presents the radioactive waste stream source term. This includes a projection of the volumes of individual waste streams anticipated to be generated from 1981 to the year 2030 (inclusive) plus projected concentrations of particular individual radionuclides. Appendix B is closely related to Appendix A and presents the waste processing options plus values associated with individual waste stream specific decision indices.

Appendix C presents four types of information. It presents the reference disposal facility considered in the report as well as the alternative disposal technologies and operating practices. It then presents the assumptions and algorithms used to determine economic costs for disposal facility siting, operation, closure, surveillance, and institutional control. Finally, it presents the assumed environmental parameters for the alternative treatment and disposal sites considered in the report.

Appendix D provides a discussion of the pathway dose conversion factors and other environmental transfer factors and numerical algorithms used to determine radiological impacts to individuals and populations. Finally, Appendix E presents a number of different types of information. This information is used as a backup to many of the assumptions for the pathway analysis formulations contained in the main body of the report.

CHAPTER 1.0 REFERENCES

(1) Oztunali, O.I., et al., "Data Base for Radioactive Waste Management," Three Volumes, Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-1759, November 1981.

(2) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Draft Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0782, September 1981.

(3) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Final Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0945, November 1982.

(4) U.S. Nuclear Regulatory Commission, "10 CFR Parts 2, 19, 20, 21, 30, 40, 51, 61, 70, 73, and 170: Licensing Requirements for Land Disposal of Radioactive Waste, Final Regulation," Federal Register, 47 FR 57446, December 27, 1982.

(5) U.S. Nuclear Regulatory Commission, "10 CFR 60: Disposal of High-Level Radioactive Wastes in Geologic Repositories, Technical Criteria," Federal Register, 48 FR 28194, June 21, 1983.

2.0 OVERVIEW OF ANALYSIS METHODOLOGY

The intent of this study is to update the impacts analysis methodology used in the preparation of the draft and final Environmental Impact Statements (EIS) (Refs. 1, 2) on 10 CFR Part 61: "Licensing Requirements for Land Disposal of Radioactive Waste" (Ref. 3). The original Part 61 analysis methodology is described in the supporting documents entitled "Data Base For Radioactive Waste Management" (Refs. 4, 5). The need for the study and the general approach was detailed in the previous chapter. This chapter presents an overview of the updated impacts analysis methodology.

For convenience, the analysis methodology is structured around two major and four minor computer codes that use a number of common data files. The information contained in the data files is a combination of generic data and user supplied information. A number of basic waste streams with generic waste processing scenarios are provided as are a number of disposal technologies that can be located in one of four generic regional environments. The code user, however, must

(1) provide information on the combination of waste streams to be considered and the regions they are generated,
(2) select the specific waste processing scenarios,
(3) select the environmental setting of the disposal site (or provide site specific data), and
(4) select the particular combination of disposal technologies to be used.

The major codes, called CLASIFY and IMPACTS, address the forward problem. IMPACTS calculates radiological impacts resulting from the implementation of the combination of alternatives selected for the above four components. The four minor codes are called INVERSE, ECONOMY, INTRUDE, and VOLUMES. The INVERSE code addresses the reverse problem, i.e., given the last two of the above four components, it calculates acceptable radionuclide total activity and/or concentration limits for disposal. The INTRUDE code analyzes radiological impacts to an inadvertant intruder as a function of time. The ECONOMY code calculates transportation and routine operational radiological impacts as well as per unit waste volume costs associated with disposal operations given annual volumes of waste disposed in six waste classes. Finally, the VOLUMES code calculates and updates region and waste stream dependent annual volume projections.

The codes can consider an arbitrary number of waste streams which may be shipped across regional boundaries, processed by the waste generator or at a centralized waste processing facility, and disposed at a location using any combination of up to six different disposal technologies. Therefore, the number of potential alternative combinations appears to be unlimited, and an overview chapter appears to be necessary which outlines the types of information processed by the codes, the available set of generic data, and the types of impacts calculated.

This overview chapter is separated into four sections. The first section outlines the scope of the calculational methodology, including the information which must be input and the impact measures which are calculated. The first section also addresses some of the limitations of the original Part 61 analysis methodology and presents a discussion on the limitations of this study. Section 2.2 describes in detail the analysis methodology used by the codes to signify the different waste characteristics, processing and disposal options, and disposal site environmental settings (i.e., the integer decision indices). Section 2.3 describes the overall approach used to calculate the release of radionuclides into the environment, their transfer through the environment, and subsequent human exposures. Finally, Section 2.4 presents a brief description of the codes. Details on the codes can be found in the second volume of this report titled "Codes and Example Problems."

2.1 Scope of the Methodology

To operate the analysis methodology computer codes, the user must provide information on radioactive waste streams and their physical, chemical, and radiological properties. The user must also specify certain waste transport parameters as well as the methodologies through which waste streams are processed and/or disposed including the environments in which processing and/or disposal takes place. This section discusses these information requirements and presents a section (Section 2.1.3) on the limitations of the results and their interpretation. But first, a brief section on the original Part 61 impacts analysis methodology appears to be indicated.

2.1.1 Original Part 61 Analysis Methodology

The original Part 61 analysis methodology was developed with a specific purpose in mind: comparative analysis of various available options on land disposal of radioactive waste in view of the wastes expected to be generated in the near-future, the commercially available waste processing and disposal technologies, and past disposal experience. The original Part 61 analysis methodology was created to aid in development of the Part 61 rule, and therefore considered a number of alternatives that were originally pertinent, but have since become moot, with the promulgation of the regulation. The four principal components of the original Part 61 analysis methodology data base include:

(1) Projected waste volumes and characteristics,
(2) Various waste processing options;
(3) Environmental properties of potential waste treatment and/or disposal locations; and
(4) Commercially available disposal technologies.

These original data bases are briefly summarized below in Sections 2.1.1.1 through 2.1.1.3. Section 2.1.1.4 presents a brief summary of the impacts analysis calculations performed, and Section 2.1.1.5 presents a discussion of some of the shortcomings of the old methodology and some improvements that were possible but, however, were not made due to time and budgetary constraints.

2.1.1.1 Waste Stream Characteristics and Processing Options

Radioactive waste exists in a very wide range of types, forms, and activities. To enable decisions regarding waste disposal to be made for the Part 61 regulation, a radioactive source term had to be developed in which this wide range was characterized in terms of physical, chemical, and radiological properties. In addition, the source term had to be structured so that several alternative waste treatment and packaging options could be considered. As discussed in the draft EIS (Ref. 1) and accompanying data base reports (Refs. 4, 5), this source term was generated by first identifying and aggregating all the various types and forms of waste expected to be generated from 1980 to the year 2000 into a manageable number of waste streams. A total of 36 waste streams were identified for the draft EIS, while a 37th waste stream was identified and added for the final EIS (Ref. 2). These waste streams are summarized in Table 2-1.

The volumes projected to be generated from 1980 to the year 2000 for each of the 37 waste streams were then determined on a regional basis. To do this, the contiguous United States was assumed to be divided into four regions which correspond to the five existing NRC regions. That is, the northeast region corresponds to NRC Region I, the southeast region corresponds to NRC Region II, the midwest region corresponds to NRC Region III, and the western region corresponds to the combined NRC Regions IV and V. The total as generated volume of each waste stream given in Table 2-1 was considered for each of the four regions. Moreover, each waste stream in each of the four regions were characterized in terms of physical, chemical, and radiological properties. Up to 23 radionuclides were considered for each waste stream.

The source term was constructed so that a large number of different waste processing and management options could be considered for each waste stream. It was recognized, however, that such flexibility would be unwieldy unless a mechanism was provided to bound the range of such options. To do this, the draft EIS considered four waste spectra, and the final EIS added two new spectra to the previous four and considered six waste spectra. Each waste spectrum denotes a cross section of all the waste streams that might be generated and shipped for disposal under specific conditions of processing and packaging. Each spectrum is defined in terms of the total waste volume, waste performance characteristics, and radionuclide concentrations that result from the application of a specific combination of waste processing options to specific waste streams. A summary of the six waste spectra considered for the final EIS is included in Table 2-2.

This detailed consideration of waste streams permitted the formulation of a set of indices, called the waste form behavior indices, that allowed some credit to be taken for waste form during the calculation of impacts. These indices are summarized in Table 2-3.

TABLE 2-1 . Waste Streams Used in the Original Part 61 Analysis Methodology

Waste Stream	Symbol
Group I: LWR Process Wastes	
PWR Ion Exchange Resins	P-IXRESIN
PWR Concentrated Liquids	P-CONCLIQ
PWR Filter Sludges	P-FSLUDGE
PWR Filter Cartridges	P-FCARTRG
BWR Ion Exchange Resins	B-IXRESIN
BWR Concentrated Liquids	B-CONCLIQ
BWR Filter Sludges	B-FSLUDGE
Group II: Trash	
PWR Compactible Trash	P-COTRASH
PWR Noncompactible Trash	P-NCTRASH
BWR Compactible Trash	B-COTRASH
BWR Noncompactible Trash	B-NCTRASH
Fuel Fabrication - Compactible Trash	F-COTRASH
- Noncompactible Trash	F-NCTRASH
Institutional Trash - Large Facilities (LF)	I-COTRASH
Institutional Trash - Small Facilities (SF)	I+COTRASH
Industrial Source and Special Nuclear Material Trash - LF	N-SSTRASH
Industrial Source and Special Nuclear Material Trash - SF	N+SSTRASH
Industrial Low Activity Trash - LF	N-LOTRASH
Industrial Low Activity Trash - SF	N+LOTRASH
Group III: Low Specific Activity Wastes	
Fuel Fabrication Process Wastes	F-PROCESS
Uranium Hexafluoride Facilities Process Wastes	U-PROCESS
Institutional Wastes - Liquid Scintillation Vials - LF	I-LIQSCVL
Liquid Scintillation Vials - SF	I+LIQSCVL
Absorbed Liquids - LF	I-ABSLIQD
Absorbed Liquids - SF	I+ABSLIQD
Biowaste - LF	I-BIOWAST
Biowaste - SF	I+BIOWAST
Industrial Source and Special Nuclear Material Waste	N-SSWASTE
Industrial Low Activity Waste	N-LOWASTE
Group IV : Special Wastes	
LWR Nonfuel Reactor Components	L-NFRCOMP
LWR Decontamination Resins	L-DECONRS
Waste From Medical Isotope Production Facilities	N-ISOPROD
Tritium Production Waste	N-TRITIUM
Accelerator Targets	N-TARGETS
Sealed Sources	N-SOURCES
High Activity Wastes	N-HIGHACT
Mixed Oxide Facility Decon Wastes	F-PUDECON

Table 2-2 . Summary Description of Original Waste Spectra

Waste Spectrum 1. This spectrum assumes a continuation of past and in some cases existing waste management practices. Some of the light water reactor (LWR) wastes are solidified; however, no processing is done on organics, combustible wastes, or streams containing chelating agents. LWR resins and filter sludges are assumed to be shipped to disposal sites in a dewatered form. LWR concentrated liquids are assumed to be concentrated in accordance with current practices, and are solidified with various media designated as solidification scenario A*. No special effort is made to compact trash. Institutional waste streams are shipped to disposal sites after they are packaged in currently utilized absorbent materials. Resins from LWR decontamination operations are solidified in a medium with highly improved characteristics (solidification scenario C*).

Waste Spectrum 2. This spectrum assumes that LWR process liquids are solidified using improved solidification techniques (solidification scenario B*). Prior to solidification, LWR concentrated liquids are additionally reduced in volume to 50 weight percent solids through use of evaporator/crystallizers. In the case of cartridge filters, the solidification agent fills the voids in the packaged wate but does not increase the volume. Liquid scintillation vials are crushed at large facilities and packed in absorbent material. All compactible trash streams are compacted, most at the source of generation and some at the disposal facility. Liquids from medical isotope production facilities are solidified using solidification scenario C procedures.

Waste Spectrum 3. In this spectrum, LWR process wastes are solidificed assuming that further improved solidification agents are used (solidification scenario C). LWR concentrated liquids are first evaporated to 50 weight percent solids. All possible incineration of combustible material (except LWR process wastes) is performed; some incineration is done at the source of generation and some at the disposal site. All incineration ash is solidified using solidification scenario C procedures.

Waste Spectrum 4. This spectrum assumes extreme volume reduction. All waste streams amenable to evaporation or incineration with fluidized bed technology are calcined and solidified using solidification scenario C procedures; LWR process wastes, except cartridge filters, are calcined in addition to the streams incinerated in waste spectrum 3. All noncombustible trash wastes are reduced in volume at the disposal site or at a central processing facility using a large hydraulic press. This spectrum represents the maximum practical level of volume reduction that can be currently achieved.

Waste Spectrum 5. This spectrum incorporates for most waste streams high integrity containers (HIC's) to achieve a stable waste form. Relative to Waste Spectrum 1, all waste streams other than activated metals which had previously been in an unstable form are stabilized using HIC's. Activated metals are stabilized by filling interstitial voids in a waste container with a noncompressible material. LWR concentrated liquids are solidified assuming solidification scenario B procedures, while waste from medical isotope production facilities is assumed to be solidified using solidification scenario C. Wastes from tritium manufacturing facilities are placed in HIC's. All compressible waste streams are compacted into HIC's most at the source of generation and some at the regional processing center assumed to be colocated with the disposal facility.

Waste Spectrum 6. This waste spectrum represents overall waste characteristics projected to result without requirements for waste stability and considering the increasing costs for waste disposal. Similarly to waste spectrum 1, most higher activity waste streams are disposed in an unstable manner. LWR resins and filter sludges are shipped in a dewatered form. Pressurized water reactor (PWR) cartridge filters, LWR non-fuel reactor core components, and LWR noncompressible trash are also packaged in a nonstable manner. LWR concentrated liquids are reduced in volume to 50 weight percent solids and are solidified. Similarly to waste spectrum 2, all compactible waste streams are compacted: some at the source of generation and some at a regional processing center assumed to be colocated with the disposal facility.

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* Solidification scenario A : 50% urea formaldehyde and 50% cement;
Solidification scenario B : 50% cement and 50% vinyl ester styrene;
Solidification scenario C : 100% vinyl ester styrene.

TABLE 2-3 . Original Waste Form Behavior Indices

Index	Parameter	Values	Explanation
I4	FLAMMABILITY	0	Non-flammable
		1	Low flammability
		2	Burns if heat supplied
		3	Flammable
I5	DISPERSIBILITY	0	Near zero
		1	Slight to moderate
		2	Moderate
		3	Severe
I6	LEACHABILITY	1	Unsolidified waste form
		2	Type A solidification
		3	Type B solidification
		4	Type C solidification
I7	CHEMICAL CONTENT	0	No chelating agents or organic chemicals
		1	Chelating agents or organic chemicals likely present
I8	STABILITY	0	Structurally unstable waste form
		1	Structurally stable waste form (inherent to waste form)
		2	Stabilized using a high integrity container
		3	Stabilized by other means
I9	ACCESSIBILITY	1	Readily accessible
		2	Moderately accessible
		3	Accessible with difficulty

It had been common practice in the past to give partial or no credit to the waste form properties in the calculation of impacts. Some credit was sometimes given to the comparative leachability of the solidification agent utilized and this effect was considered in groundwater impact calculations. However, a quantitative analysis of the mechanical strength, thermal properties, resistance to chemical and biological attack, resistance to leaching, and other properties of the waste form and their effects on all the pathways considered had not been performed.

The primary reason for this past conservatism was the lack of detailed data on the different types of wastes included in the impact analyses. All the LWR wastes or all the non-fuel cycle wastes, or both, were considered as one stream. A contributing reason for this conservatism was the lack of data on the performance of wastes over long periods of time. However, in the original Part 61 methodology, as well as this report, the waste is separated into many individual waste streams and each stream is considered separately in the impact calculations. Consequently, wide variations in waste stream properties can be quantified based on available quantitative, qualitative, and comparative data.

2.1.1.2 Reference Environments

It is apparent that the environmental characteristics of the location at which waste processing or disposal takes place will influence the exposures calculated. For example, if a waste stream is processed through incineration, then radionuclides may be released into the air, and the resulting calculated impacts will be influenced by factors such as the average wind speed at the processing site, wind stability classes, and the distribution and density of the surrounding population. As another example, groundwater migration of radionuclides from a disposal facility site will be influenced by a number of considerations including the amount of rainfall, the ion-exchange properties of the site soil, and the distance from the site to a point of discharge such as a well.

It is clearly impossible to consider all possible variations in processing or disposal site environmental characteristics. As a compromise, four reference treatment and/or disposal facility site environments were formulated for the original Part 61 analysis methodology. Each of the four reference disposal site locations is located within a region which corresponds to the waste generating regions discussed above, and could represent a large multi-state compact formed for waste disposal. Qualitative descriptions of the reference environments are given in Table 2-4.

The regional environmental properties are meant to be typical of environmental characteristics of the regions (not necessarily the "best" site that could be located within a region) and were developed from a number of sources. Thus, these properties should not be interpreted as representing existing or possibly planned disposal facility, or any specific location within a region.

Two additional variations of the southeast site were considered to enable illustration of some specific points in the Part 61 draft and final EIS.

TABLE 2-4 . Qualitative Description of Reference Site Environments

Region Index	Site Name	General Environment	Population Density	Soil Permeability
1	Northeast	Humid	High	Low
2	Southeast	Humid	Moderate	Moderate
3	Midwest	Humid	Low	Low
4	Southwest	Semi-Arid	Low	High

One variation assumed significantly lower permeability site soils, while the other variation assumed significantly higher permeability site soils.

All reference site locations were assumed to be located in rural environments. The codes, however, allowed the code user to consider airborne radiological impacts resulting from incineration of waste. For some facilities such as a hospital or a university, waste processing could take place in an urban setting. The codes, therefore, allowed adjustment of the assumed population densities around the reference sites to more accurately reflect an urban environment.

2.1.1.3 Disposal Technologies

The potential number of available disposal technologies and operational practices are as numerous as the number of available waste processing options. In a manner similar to the approach adopted in the grouping of an unwieldy number of waste processing alternatives into a few manageable options, the available disposal alternatives were also grouped into a few significant options.

The disposal technologies considered in detail in the original analysis methodology included a shallow earth trench, a concrete walled trench, and a "hot waste" facility (a small concrete trench). Operational variations considered included options on (1) waste segregation (none, stability based, and organic chemical based), (2) waste emplacement (random, stacked, and decontainerized), (3) compaction of disposal cells (minimal, moderate, and extreme), (4) backfill (onsite soil, sand, and grout), and (5) cover of disposal cells (minimum, improved, and intruder barrier). These options were expressed through several integer decision indices that referred to a specific procedure used in the impact analysis methodology or determined a specific value associated with disposal design. Table 2-5 presents a summary of these original decision indices.

2.1.1.4 Impact Measures and Analysis Methodology

The original Part 61 analysis methodology utilized five impact measures: costs, land use, energy use, occupational exposures, and public exposures including those to individuals and to populations.

TABLE 2-5 . Original Disposal Technology Indices

Index	Parameter	Values	Explanation
IR	REGION	1-6	Various regional sites (see text)
ID	DESIGN	1	Regular shallow land burial trenches
		2	"Concrete-walled" trenches
IC	COVER	1	Regular cover
		2	"Thick" cover
		3	"Intruder barrier" cover
IX	STABILIZATION	1	No special compaction
		2	Moderately extensive compaction
		3	Very extensive compaction
IE	EMPLACEMENT	1	Random
		2	Stacked
		3	Decontainerized
		4	Random with sand backfill
		5	Stacked with sand backfill
IS	SEGREGATION	0	No segregation
		1	Segregation of unstable waste
		2	Segregation of waste containing chemicals
		3	Segregation of unstable waste as well as waste containing chemicals
IL	LAYERING	0	No layering
		1	Layering of waste streams
IG	GROUTING	0	No grouting
		1	Grouting of spaces between disposed packages
IH	HOT WASTE FACILITY	0	No special disposal of high-activity waste
		1	Special disposal for high-activity waste
IQ	CLOSURE	1	2 year modest closure effort
		2	4 year closure period incorporating complete site restabilization program
ICL	CARE LEVEL	1	Low care level
		2	Moderate care level
		3	High care level
IPO	POST-OPERATIONAL PERIOD	2-99*	Number of years between cessation of disposal of waste and transfer of title to site owner
IIC	INSTITUTIONAL CONTROL PERIOD	0-999	Number of years between transfer of title to site owner and the assumed loss of institutional controls

Impacts calculated for waste processing included costs, energy use, occupational exposures, and population exposures. For costs, energy use, and occupational exposures, unit impact measures (e.g., per unit volume of waste processed) were used. These unit impact measures were based on a number of sources and were assumed to be adequate for comparison purposes -- that is, the calculated impacts did not necessarily present "realistic" values. Among the processing options, only incineration was considered to result in population exposures, and impacts were calculated based on the above reference environmental settings and assumed radionuclide-specific fractions of activity released which varied depending on which of three types of incinerators were considered.

Transportation impacts calculated included costs, energy use, population exposures during transit, and occupational exposures to waste generator personnel (loading), truck drivers, and disposal facility personnel (unloading and emplacement).

Both short-term and long-term impacts were calculated for disposal. The short-term impacts included costs, energy use, occupational impacts due to routine operations and operational accidents. The methodology was later improved to consider public exposures which could result from leachate accumulation (Ref. 2). The long-term impacts calculated included costs, energy use, and public exposures from a multitude of scenarios including inadvertant intrusion (two scenarios at three different times), groundwater (four different scenarios at several different times), and exposed waste (four scenarios).

2.1.1.5 Discussion

In retrospect, there appears to exist several places where improvements could be made in the original Part 61 analysis methodology, as well as several limitations. In this context, an improvement means a possible refinement of the data base and/or calculational methodology. A limitation is a shortcoming in the data base and/or calculational methodology which was acknowledged at the time, but was deemed to not affect the adequacy of the methodology as an input to rulemaking. These improvements and limitations are discussed below.

Some of the improvements which could be made to the original Part 61 analysis methodology include:

(1) A more detailed treatment of ground water migration by considering unsaturated zone transport separate from saturated zone transport. In the original analysis methodology, flow through each of these two regions was combined to form a single radionuclide-specific travel time to a given biota access location.

(2) A more comprehensive list of radionuclides. Only 23 radionuclides were originally considered, whereas many more radionucides are present in the various radioactive waste streams.

(3) Calculation of radon emanation and ingrowth within dwellings as part of determining impacts due to potential inadvertent intrusion into disposed waste.

(4) An update of the health physics methodology used to determine impacts from radioactive contamination at a given biota access location. This could include updating radionuclide uptake and transfer factors to reflect regional dependency and acquisition of additional technical data, as well as updating organ-specific dose conversion factors to reflect advancement in biological models for ingestion, inhalation, and direct exposure.

(5) A more detailed treatment of highly engineered disposal facilities such as concrete bunkers.

(6) A more detailed treatment of the potential impacts from disposal of certain waste forms which are generated infrequently, tend to be physically very small (e.g., 1 ft^3 or less), but also tend to be highly concentrated. These include some activated metals, in which potential releases can be directly related to the metal corrosion rate, and sealed sources. These waste streams appeared to require unique algorithms for calculation of radiological impacts. However, the volumes of these waste streams were projected to be exceedingly small, and the effort was deemed to be unjustified at the time.

(7) A more detailed consideration of the factors that contribute to calculation of disposal facility costs. This includes costs during preoperational siting, licensing, and construction, as well as operational costs.

(8) A more accurate analysis of transportation impacts which more fully incorporates the radiological characteristics of specific waste streams. The original Part 61 analysis methodology used a simple approximation based on WASH-1238 (Ref. 6).

(9) Different solubilities for the same radionuclide could not be considered in the impacts calculations.

(10) A more detailed treatment of waste streams routinely expected for disposal. This includes identification of additional important waste streams and more precise characterization of existing waste streams.

Some of the limitations of the original Part 61 analysis methodology are as follows:

(1) It could not easily consider use of several different disposal technologies at a single disposal facility site. An example would be the use of shallow trenches for Class A waste and concrete bunkers for Class B and C waste.

(2) The analysis methodology was principally geared to large disposal facilities (e.g., 50,000 m^3 of waste per year and could not easily consider disposal costs for facilities possibly accepting a wide range in annual waste volumes. This comment is also applicable to the calculational algorithms used for calculating impacts from ground water migration.

(3) It did not include several potential future waste streams from such sources as nuclear fuel reprocessing or decontamination and/or decommissioning of nuclear power reactors.

(4) For the intruder well scenario, the only impacts which were considered were those from potential consumption and use of contaminated water. Well drilling, however, could potentially intersect the

disposed waste, bringing radioactive contamination to the surface which could impact a potential inadvertent intruder through other pathways.

(5) Only intraregional transportation and disposal was considered. This was consistent with the then evolving state compacting process. However, cross-regional waste shipment still occurs today and may continue to do so in the future for some waste streams.

The reason for listing these particular improvements and limitations is because they have been remedied in this report. Certain other limitations remain from the original Part 61 analysis methodology, however, and they are discussed in Section 2.1.3.

2.1.2 Modifications to the Original Part 61 Analysis Methodology

The changes made to the original Part 61 analysis methodology were briefly summarized in Section 2.1.1.5. This section further discusses these changes in the context of modifications to the data bases as well as some of the calculational algorithms.

As mentioned in the previous section, the analyses in this report are based on the following four data base components:

(1) Radioactive wastes expected to be generated in the near future;
(2) Available waste processing options;
(3) Environmental properties of the reference locations at which these wastes are treated and disposed; and
(4) Alternative disposal technologies that may be used to dispose of these wastes.

All of these components have been modified from the data base used in the original Part 61 analysis methodology. However, the most important modification is the existence of 10 CFR Part 61 (Ref. 3). This rule creates an overall framework for the assumptions on the above components. For example, Part 61 defines three classes of waste with minimum waste form requirements and radionuclide concentration upper limits for each class. Similarly there are minimum acceptability requirements for siting of disposal facilities, and guidelines for their design, operation, and closure.

Sections 2.1.2.1 through 2.1.2.3 summarize the modifications to the data bases, and Section 2.1.2.4 briefly discusses the modifications to the calculational algorithms and impact measures.

2.1.2.1 Waste Stream Characteristics and Processing Options

The most apparent change is in the waste stream data base. Most of the original 37 waste streams presented in Table 2-1 have been kept, although several improvements have been made in the projected volumes, and physical and radiological characteristics. Many additional waste streams have also been characterized. These additional waste streams can be grouped into three types: (1) routine wastes of small volumes and relatively high concentrations (e.g., waste from some sealed source manufacturers), (2) wastes that depend on formulation and implementation of certain decisions (e.g., reprocessing of spent-fuel, and decommissioning through dismantlement of nuclear power plants, and (3) other wastes (e.g., waste from the West Valley Demonstration Project). The waste streams considered in this report are described in Table 2-6. The following presents significant changes in the waste stream data base:

(1) New and more accurate volume generation rates are used for LWR waste streams based on recent survey data. This new set of volume generation rates permits consideration of LWRs in five categories: PWR Fresh Water Site (FWS), PWR Salt Water Site (SWS), BWR-FWS with Filter Demineralizers, BWR-FWS with Deep Bed Demineralizers, and BWR-SWS.
(2) An activity distribution has been obtained based on shipped waste data and assumed for the four trash waste streams from LWRs.
(3) N-SOURCES waste stream has been split into several other waste streams, and it is characterized much more accurately.
(4) N-ISOPROD and N-TRITIUM waste streams are also characterized in more detail than previously;
(5) Waste streams containing radium-226, including sealed sources and ion exchange resins from groundwater treatment processes, have been included.
(6) Wastes from some military establishments, small tritium manufacturers, and large sealed source manufacturers have been characterized and added to the data base.
(7) A large number of waste streams have been characterized from potential nuclear fuel cycle operations, including reprocessing waste and waste from mixed oxide fuel fabrication facilities.
(8) A large number of waste streams from future decommissioning of LWRs have been included, as have wastes from the West Valley Demonstration Project.
(9) Activated metal wastes (e.g., end pieces and spacers) from potential fuel consolidation activities have been included.

In addition, as pointed out above, the 23 radionuclides considered in the original Part 61 analysis methodology was considered to be relatively limiting. This report considers 53 radionuclides. However, many of these radionuclides have more than one solubility class. Consequently, this report considers a total of 100 distinct radionuclide/solubility combinations. These are presented in Table 2-7.

Waste form behavior indices are basically similar to those defined in Table 2-3. However, several of the indices have been expanded and are used for more decisions than those defined in Table 2-3. These definitions are detailed in Section 2.2.

Finally, instead of the six waste spectra considered in the original Part 61 analysis methodology, five waste spectra are considered in this report. These waste spectra are generally similar to those defined in Table 2-2, although some significant changes have been made. The most significant changes include consolidation of the old waste spectra 1 and 2, consideration of improved waste solidification process control programs, and

TABLE 2-6 . Waste Groups and Streams

I. Nuclear Power Plants	Symbol	No.
PWR ion-exchange resins	P-IXRESIN	1
PWR concentrated liquids	P-CONCLIQ	2
PWR filter sludges	P-FSLUDGE	3
PWR cartridge filters	P-FCARTRG	4
BWR ion-exchange resins	B-IXRESIN	5
BWR concentrated liquids	B-CONCLIQ	6
BWR filter sludges	B-FSLUDGE	7
PWR combustible trash	P-COTRASH	8
PWR noncombustible trash	P-NCTRASH	9
BWR combustible trash	B-COTRASH	10
BWR noncombustible trash	B-NCTRASH	11
LWR nonfuel reactor core components	L-NFRCOMP	12
LWR decontamination waste	L-DECONRS	13
II. Other Nuclear Fuel Cycle Facilities		
Fuel fabrication process waste	F-PROCESS	14
Fuel fabrication combustible trash	F-COTRASH	15
Fuel fabrication noncomb trash	F-NCTRASH	16
UF6 conversion process waste	U-PROCESS	17
Pu facility decontamination waste	L-PUDECON	18
Waste from hot cell burnup studies	L-BURNUPS	19
III. Institutional Waste		
Combustible trash (LF)[a]	I-COTRASH	20
Combustible trash (SF)[b]	I+COTRASH	21
Absorbed liquids (LF)	I-ABSLIQD	22
Absorbed liquids (SF)	I+ABSLIQD	23
Liquid scintillation vial waste (LF)	I-LIQSCVL	24
Liquid scintillation vial waste (SF)	I+LIQSCVL	25
Biological waste (LF)	I-BIOWAST	26
Biological waste (SF)	I+BIOWAST	27
IV. Industrial Waste		
Source and SNM trash (LF)	N-SSTRASH	28
Source and SNM trash (SF)	N+SSTRASH	29
Source and SNM waste	N-SSWASTE	30
Low activity trash (LF)	N-LOTRASH	31
Low activity trash (SF)	N+LOTRASH	32
Low activity waste	N-LOWASTE	33
Large Radioisotope Manufacturers		
High-activity production trash	N-ISOPROD	34
Low-activity production trash	N-ISOTRSH	35
Large sealed source manufacturers	N-SORMFG1	36
	N-SORMFG2	37
	N-SORMFG3	38
	N-SORMFG4	39

IV. Industrial Waste (cont'd)	Symbol	No.
Large Tritium and C-14 Manufacturers		
Compactible trash	N-NECOTRA	40
Absorbed organic liquids	N-NEABLIQ	41
Solidified aqueous liquids	N-NESOLIQ	42
Rejected product vials	N-NEVIALS	43
Non-compactible glass	N-NENCGLS	44
Non-compactible wood and metal	N-NEWOTAL	45
Tritium gas	N-NETRGAS	46
Absorbed tritiated liquid	N-NETRILI	47
Absorbed C-14 liquids	N-NECARLI	48
Laboratory trash	N-MWTRASH	49
Absorbed organic liquids	N-MWABLIQ	50
Solidified aqueous liquids	N-MWSOLIQ	51
Miscellaneous waste	N-MWWASTE	52
Small Tritium Manufacturers		
Tritium in paint or as plating	N-TRIPLAT	53
Gaseous tritium	N-TRITGAS	54
High-activity scintillation liquids	N-TRISCNT	55
Tritium in aqueous liquid	N-TRILIQD	56
Miscellaneous trash	N-TRITRSH	57
Tritium cont./absorbed in metal	N-TRIFOIL	58
High activity waste	N-HIGHACT	59
Sealed sources and devices		
Tritium sources	N-TRITSOR	60
Carbon-14 sources	N-CARBSOR	61
Cobalt-60 sources	N-COBLSOR	62
Nickel-63 sources	N-NICKSOR	63
Strontium-90 sources	N-STROSOR	64
Cesium-137 sources	N-CESISOR	65
Plutonium-238 sources	N-PLU8SOR	66
Plutonium-239 sources	N-PLU9SOR	67
Americium-241 sources	N-AMERSOR	68
Pu-238 neutron sources	N-PUBESOR	69
Am-241 neutron sources	N-AMBESOR	70
V. Other Non-Fuel Cycle Waste		
Radium sources		
Medical needles	N-RANEEDS	71
Medical cells	N-RACELLS	72
Medical plaques	N-RAPLAQU	73
Medical nasopharyngeal applicators	N-RANPAPP	74
Radium-beryllium neutron sources	N-RABESOR	75
Miscellaneous non-medical sources	N-RAMISCL	76
Radium ion-exchange resins	N-RARESIN	77
Navy dry waste	M-NAVYDRY	78
Navy wet waste	M-NAVYWET	79

TABLE 2-6 . Waste Groups and Streams (cont'd)

VI. Non-Routine Waste	Symbol	No.
Uranium Fuel Processing		
High-level liquid waste	R-HLLWFRP	80
Fuel assembly hardware	R-FUEHARD	81
Hulls from chop/leach process	R-HULLFRP	82
Intermediate-level liquid waste	R-ILLWFRP	83
Silica gel	R-SILIGEL	84
Main plant high-activity comp trash	R-MPCOTRH	85
Main plant low-activity comp trash	R-MPCOTRL	86
Main plant noncompressible trash	R-MPNCTRA	87
Degraded extractant	R-DEGREXT	88
Main plant ion-exchange resins	R-MPRESIN	89
Storage basin resin and filter sludge	R-SBRESIN	90
Storage basin concentrated liquids	R-SBCOLIQ	91
Storage basin compressible trash	R-SBCOTRA	92
Storage basin noncompressible trash	R-SBNCTRA	93
UF_6 conv flourinator residues	R-UFFINES	94
UF_6 conv K_2UO_4 mud	R-UFK2MUD	95
UF_6 conv compressible trash	R-UFCOTRA	96
UF_6 conv noncompressible trash	R-UFNCTRA	97
PuO_2 conv compressible trash	R-PUCOTRA	98
PuO_2 conv noncompressible trash	R-PUNCTRA	99
Mixed Oxide Fuel Fabrication		
Compressible trash	R-MOXCOTR	100
Noncompressible trash	R-MOXNCTR	101
Proc & scrap recovery solutions	R-MOXSOLN	102
Nuclear Power Plant Decommissioning		
PWR activated core shroud	P-DECORES	103
PWR activated reactor internals	P-DEACINT	104
PWR activated reactor vessel	P-DEACVES	105
PWR activated concrete	P-DEACTCO	106
PWR contaminated metals	P-DECONME	107
PWR contaminated concrete	P-DECONCO	108
PWR combustible/compactible trash	P-DETRASH	109
PWR chelated ion-exchange resins	P-DERESIN	110
PWR filter cartridges	P-DEFILCR	111
PWR evaporator bottoms	P-DEEVAPB	112
BWR activated core shroud	B-DECORES	113
BWR activated reactor internals	B-DEACINT	114
BWR activated reactor vessel	B-DEACVES	115
Nuclear Power Plant Decommissioning (cont'd)		
BWR activated concrete	B-DEACTCO	116
BWR contaminated metals	B-DECONME	117
BWR contaminated concrete	B-DECONCO	118
BWR combustible/compactible trash	B-DETRASH	119
BWR chelated ion-exchange resins	B-DERESIN	120
BWR evaporator bottoms	B-DEEVAPB	121
West Valley Demonstration Project		
Thorex high-level waste	W-THORHLW	122
Purex high-level waste	W-PUREHLW	123
Trash from existing systems	W-COTRASH	124
Miscellaneous dry solids	W-NCSOLID	125
LLWTF sludge and resins	W-LLWTFRE	126
FRS filter precoat and resins	W-FRSRESN	127
RTS liquid waste	W-FRSLIQD	128
RTS filter backwash and resins	W-RTSRESN	129
Trash, low TRU content	W-LTTRASH	130
Trash, high TRU content	W-HTTRASH	131
Equipment and hardware, low TRU	W-LTEQUIP	132
Equipment and hardware, high TRU	W-HTEQUIP	133
PD liquid waste	W-PDWLIQD	134
Vitrification Waste		
Supernate	W-VITSUPR	135
Sludge wash	W-VITWASH	136
Scrub condensate	W-VITSCRB	137
Melter feed overheads	W-VITMELT	138
Fractionator condensate	W-VITFRAC	139
Zeolite slurry	W-VITZEOL	140
Decontamination & Decommission		
Fuel storage racks	W-DDRACKS	141
Low TRU rubble	W-DDLTRUB	142
High TRU rubble	W-DDHTRUB	143
Low TRU liquids	W-DDLTLQD	144
High TRU liquids	W-DDHTLQD	145
Resins	W-DDRESIN	146
VII. Other Waste		
Spent nuclear power plant fuel	L-SPENTFU	147
Power plant fuel assembly hardware	L-FUEHARD	148

Note : a - Large Facility; b - Small Facility.

TABLE 2-7 . Radionuclides Considered

Nuclide	Solubilities	Half-Life (Years)	Nuclide	Solubilities	Half-Life (Years)
H-3	*[1]	1.23E+01[2]	Th-228	W,Y	1.91E+00
C-14	*	5.73E+03	Th-229	W,Y	7.34E+03
Na-22	D	2.62E+00	Th-230	W,Y	8.00E+04
Cl-36	D,W	3.08E+05	Th-232	W,Y	1.41E+10
Fe-55	W,Y	2.60E+00	Pa-231	W,Y	3.25E+04
Co-60	W,Y	5.26E+00	U-232	D,W,Y	7.20E+01
Ni-59	D,W	8.00E+04	U-233	D,W,Y	1.62E+05
Ni-63	D,W	9.20E+01	U-234	D,W,Y	2.47E+05
Sr-90	D,Y	2.81E+01	U-235	D,W,Y	7.10E+08
Nb-94	W,Y	2.00E+04	U-236	D,W,Y	2.39E+07
Tc-99	D,W	2.12E+05	U-238	D,W,Y	4.51E+09
Ru-106	Y	1.01E+00	Np-237	W,Y	2.14E+06
Ag-108m	D,W,Y	1.27E+02	Pu-236	W,Y	2.85E+00
Cd-109	D,W,Y	1.24E+00	Pu-238	W,Y	8.64E+01
Sn-126	D,W	1.05E+05	Pu-239	W,Y	2.44E+04
Sb-125	D,W	2.71E+00	Pu-240	W,Y	6.58E+03
I-129	D	1.17E+07	Pu-241	W,Y	1.32E+01
Cs-134	D	2.05E+00	Pu-242	W,Y	3.79E+05
Cs-135	D	3.00E+06	Pu-244	W,Y	7.60E+07
Cs-137	D	3.00E+01	Am-241	W,Y	4.58E+02
Eu-152	W	1.27E+01	Am-243	W,Y	7.95E+03
Eu-154	W	1.60E+01	Cm-242	W,Y	4.45E-01
Pb-210	W	2.04E+01	Cm-243	W,Y	3.20E+01
Rn-222	*	1.05E-02	Cm-244	W,Y	1.76E+01
Ra-226	W	1.60E+03	Cm-248	W,Y	4.70E+05
Ra-228	W	6.70E+00	Cf-252	W,Y	2.65E+00
Ac-227	W,Y	2.16E+01			

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(1) Solubility: * - Not applicable, D - Day, W - Week, Y - Year
(2) Exponential Notation: 1.23E+01 = 1.23 x 10^1

additional treatment/packaging options for source waste streams. The new waste spectra are summarized in Table 2-8.

2.1.2.2 Reference Environments

The four reference site environments defined in the original Part 61 analysis methodology have been preserved substantially in the same format. However, a few new parameters have been added, e.g., parameters on unsaturated zones of the reference sites. The most significant change has been to incorporate all the site related information, including transportation information, into a separate data file. This permits the user of the codes to easily alter these data and/or add site-specific information. The values of the environmental parameters for the four disposal facility sites are presented in Table 2-9.

2.1.2.3 Disposal Technologies

This component of the data base underwent a drastic change. The revised analysis methodology permits the use of up to six different disposal technologies at the same location. The number six was selected based on the assumption that each different class of waste could conceivably be disposed using a different disposal technology. Four of these six classes of waste are as follows: Class A, Class A Stable, Class B, and Class C. In addition, a hypothetical "Class D" has been included consisting of wastes that exceed Class C concentration limitations. These are separated into two subclasses for convenience: Class D1, and Class D2. Class D1 denotes waste with activities exceeding Class C limits that are routinely generated in relatively small volumes, and Class D2 denotes waste with activities exceeding Class C limits that are generated intermittently in relatively large volumes. In this report, the former (D1) is assumed to consist of Class D waste streams originating from Groups I through V in Table 2-6, while the latter (D2) is assumed to consist of Class D waste streams originating from Groups VI or VII.

All waste streams are assumed to comply with the minimum waste form requirements of 10 CFR 61.56(a) (Ref. 3), and all except Class A are assumed to comply with the stability requirements outlined in 10 CFR 61.56(b) and the Low-Level Waste Licensing Branch Technical Position on Waste Form (Ref. 7).

A number of generic disposal technologies have been characterized through the use of the disposal technology indices and associated parameters discussed in Section 2.2. A brief list of the available reference disposal technologies and operational options considered in quantitative detail in this report is presented in Table 2-10.

Any one of the disposal technologies given in Table 2-10 can theoretically be used to dispose any of the above six categories of waste. (Clearly, some disposal technologies will be incompatible with certain classes of waste, e.g., Class C waste must be disposed in accordance with the intruder protection requirements of 10 CFR 61.52(a)(2).) Moreover, the six classes of waste can potentially be mixed among each other (e.g., Class A

TABLE 2-8 . Summary Description of Waste Spectra

Waste Spectrum 1. This waste spectrum represents a continuation of past or existing waste management practices, and presents waste characteristics projected to result without requirements for waste stability. Some waste streams such as LWR concentrated liquids are solidified using solidification scenario A, although the process control programs for the waste forms are only oriented toward meeting a free standing solid requirement rather than a structural stability requirement. Fuel cycle compressible trash waste streams are compacted, as is trash from large institutional facilities. Ion-exchange resins and filter media are shipped to disposal facilities in a dewatered form. Activated metals are packaged into containers with the interstitial spaces in the container either left as voids or filled with compressible waste forms. This results in an unstable waste form.

Waste Spectrum 2. This waste spectum is similar to waste spectrum 1, although three changes are especially significant. First, liquids which were originally solidified in waste spectrum 1 are again solidified using solidification scenario A. However, solidification is generally performed using a process control program that is in accordance with the NRC Technical Position on Waste Form, or by some other method that complies with 10 CFR Section 61.56. Second, ion-exchange resins and filter sludges are solidified rather than being disposed in a dewatered form. Third, activated metal wastes are disposed in a stable manner by filling interstitial spaces within waste containers with an inert low compressible material such as sand. Sealed sources are stabilized by solidification within a drum.

Waste Spectrum 3. This waste spectrum is very similar to waste spectrum 2 except that an improved solidification media is assumed to be used (solidification scenario B) and increased compaction is performed on compressible waste. Liquids generated by large facilities are also generally subjected to increased volume reduction. The solidification process control program is generally such that the solidified waste form meets the Part 61 stability requirements. Compaction at large facilities is assumed to be generally accomplished using improved compactor/shredders. Compressible waste from a number of small facilities -- particularly small institutional and industrial facilities -- are generally compacted at a centralized operation assumed to be operated as an adjunct to the disposal facility.

Waste Spectrum 4. This waste spectrum is devised assuming extreme volume reduction. All wastes amenable to evaporation or incineration with fluidized bed technology (e.g., LWR process wastes) are calcined and then solidified using solidification scenario C. Institutional and industrial waste streams at large facilities are incinerated using a pathological incinerator, while several institutional and industrial waste streams are shipped to a large fluidized bed incinerator assumed to be located at the disposal facility. Several noncompactable and contaminated metal waste streams are also shipped to the disposal facilities where they are compacted using a large hydraulic press.

Waste Spectrum 5. This waste spectrum incorporates for most waste streams high integrity containers (HICs) to achieve a stable waste form. Relative to waste spectrum 1, most waste streams (other than activated metals) which had previously been in an unstable form are stabilized using HICs. Activated metals are stabilized by filling interstitial voids in a waste container with a noncompressible material. Concentrated aqueous liquids are solidified assuming solidification scenario A procedures and a process control program compatible with the NRC Technical Position on Waste Form. Wastes from tritium and carbon-14 manufacturing facilities are also placed into HICs, as are sealed sources.

TABLE 2-9 . Environmental Parameters Assumed

	Reference Sites*			
Environmental property	NE(1)	SE(2)	MW(3)	SW(4)
Mean average temperature °C (°F)	8°C (46°F)	17°C (63°F)	11°C (51°F)	14°C (57°F)
Average wind speed m/sec (mph)	4.61 (10.3)	3.61 (8.1)	4.72 (10.6)	6.67 (15.5)
No. of days per year having at least 0.01 in precipitation	146	115	110	65
Average annual precipitation mm (in)	1,034 (41)	1,168 (46)	777 (30.5)	485 (19)
Average annual natural percolation into groundwater system, mm(in)	75 (2.9)	180 (7.1)	50 (2.0)	1 (.04)
Precipitation-evaporation (PE) index of site vicinity	136	91	93	21
Average silt content of site soils (%)	65	50	85	65
Average cation exchange capacity (meq/100g)	15	10	12	5
Travel time (yrs), waste to water table	0	10	70	276
Groundwater speed (m/yr)	0.1	1.25	0.66	10

* The numbers in parentheses denote values for the region index, IR, which are used in the codes to specify environmental parameters for each of the reference sites (e.g., if IR = 1, then environmental parameters assumed for the northeast site are used in the calculations).

TABLE 2-10 . Disposal Technology Options Considered

	Regular Concepts				Concrete Concepts		
Options	Large Trench	Small Trench	Unlined Auger	Slit Trench	Trench	Slit Trench	Modular Repackaged
Location							
Humid	X	X	X	X	X	X	X
Arid	X	X	X	X	X	X	X
Cover							
Reference	X	X	X	X	X	X	X
Improved	X	X	X	X	X	X	X
Compaction							
Regular	X	X	X	X	X	X	X
Moderate	X	X	X	X	X	X	X
Extreme	X	X		X			
Backfill							
Soil	X	X	X	X	X	X	X
Sand	X	X	X	X	X	X	X
Grout	X	X	X	X	X	X	X
Emplacement							
Random	X	X					
Stacked	X	X	X	X	X	X	X

Stable mixed with Class B), resulting in use of fewer than six different disposal technologies. Some disposal technologies may be appropriate for disposal of all the waste.

2.1.2.4 Impact Measures and Analysis Methodology

All the impact measures used in the original Part 61 analysis methodology have been kept with one exception. Energy use is omitted in this report because it was found to be redundant. It turned out that costs were a better measure of energy use than the calculated values which were based on insufficient information and questionable assumptions.

Most of the changes made to the calculational algorithms of the impacts analysis methodology have been outlined in Section 2.1.1.5. In addition, however, one significant calculational change has been made which both expands and updates the methodology. This change concerns the manner in which waste in very small volumes with very high concentrations (e.g., sealed sources) can be classified and their impacts calculated. The original Part 61 analysis methodology was oriented towards calculation of impacts based on radioactive concentrations, i.e., activity per unit volume or mass. This treatment was adequate and sufficient to address the major portion of the waste stream volumes being generated. Since then, however, there is some interest in more precisely analyzing disposal impacts from sealed sources and other small volume waste streams. Consequently, this report allows an option to be considered for these waste streams (henceforth, generically referred to as sources) that permits their classification on a total activity per source basis. Impact assessment calculations have also been modified to allow consideration of these and other unique wastes such as activated metals. The details of these modifications are discussed in Chapter 4 and Volume 2 of this report.

2.1.3 Limitations

As in all impact analysis methodologies which attempt to describe real world events using idealized models and other simulation tools, the updated Part 61 analysis methodology still contains a number of limitations. Some of these are inherent to its main purpose in being: analysis of alternatives for rulemaking. Certain simplifying assumptions are necessary when performing a comparative analysis of alternatives which would not be as appropriate when performing an analyis of a particular site and a particular disposal design. Similarly, a methodology constructed for analysis of a specific site and facility design would contain features which would be inappropriate for a generic analysis.

Other limitations arise from the fact that the potential variables that could be envisioned are too large to model. There are approximately 20,000 licensees that could potentially generate radioactive waste, and it would be difficult and counterproductive to try and model for each licensee the precise volumes and physical, chemical, and radiological characteristics of specific waste types. Nor would it be an easy task to precisely model all impacts from waste processing and transport to the disposal facility, given the great diversity in processing options, transport routes and distances, and impacted individuals and populations. Finally, there are an uncountable number of possible disposal methods, designs and environmental settings. As a compromise against the above difficulties, a number of simplifying assumptions are made. These are discussed below.

(1) The wide and diverse spectrum of low-level wastes are grouped into a number of individual waste streams, and regional projections are made regarding volumes, physical, chemical, and radiological characteristics. Although this is a drastic simplification of the real world, it still is far superior in detail to what has been frequently done in the past. It may also be noted that some waste streams may be more readily predictable than others. For example, waste streams from a nuclear power reactor are predictable since such wastes are routinely generated as part of reactor operations. On the other hand, consider the holder of a large sealed source. This licensee generates no waste; waste in the form of the sealed source is only created when a decision is made that the source can or will no longer be used.

(2) The regional projection is still made in terms of four very large regions, which is necessary considering the time and resources necessary to complete the project. The compacts that will eventually be established, however, will probably be more numerous and considerably smaller -- perhaps comprising only one or two states. It would therefore have been preferable, if time and resources were available, to project waste stream generation and characteristics on a state-by-state basis. The waste steams from any number of different states could then be considered. As a compromise, the revised analysis methodology allows the user to consider fractional multiples of waste stream generation in order to arrive at a volume appropriate for any size region.

(3) It is impossible to consider for all licensees the costs and other impacts for waste generation and packaging, since this would require detailed knowledge of all licensees' activities. Nor is it possible to consider all possible waste processing technologies. As a compromise, then, the analysis methodology considers costs and other impacts that would arise as an increment to a "base" level of processing. Incremental processing costs include improved compaction and incineration. Postulated processing techniques are meant to be representative rather than factual and are selected from a range of possible designs.

(4) Waste transportation impacts, as well as operational impacts from waste emplacement, are calculated in a simple manner. Thus, the impact measures thus determined should be regarded as being useful on a comparative basis rather than a "realistic" basis.

(5) A representative range of near-surface disposal facility designs and operating variations are considered in detail. These represent only a few of the possible techniques, however, and so to expand the flexibility of the analysis methodology, the codes allow the user to readily alter facility parameters to model specific cases.

These limitations, however, are not deemed to be of major significance within the generic framework of this report.

2.2 Decision Indices

As pointed out in Section 2.1, the original Part 61 analysis methodology made extensive use of decision indices which were used to quantify some of the different properties of several major components of the disposal system (waste characteristics, waste treatment and processing parameters, facility site environments, and disposal technologies). This report also makes extensive use of decision indices. This section defines these decision indices, outlines their meaning, and presents alternative values.

Decision indices are usually integer values which refer to a specific procedure used in the calculation of impacts, determine a specific value of an environmental parameter, or signify specific engineering properties of a disposal technology. They are especially useful in cases where there exist a considerable number of alternative disposal technologies and/or alternative property values for a given component. All the decision indices are user-input values which are used at different points in the impacts analysis algorithms.

In this report, three different types of decision indices are used to describe the available options. Most of the decision indices are input into the codes through the files called IMPCON.DAT, CLACON.DAT, WASCAR.DAT, and DISTEC.DAT. In addition, some parameter values associated with the decision indices are supplied through several other data files. Data file structures are briefly discussed in Section 2.4 of this report, and are detailed in Volume 2.

The first type of indices consists of general facility and schedule indices that apply to the specific problem being considered. They are independent of the particular waste stream or waste class being considered and describe properties related to the disposal environment, disposal facility lifetime and establishment schedule, cost calculations, etc.

The second type of indices is waste stream specific. There are three groups of waste stream specific indices: the first group refers to the processing options (waste spectra) available for that waste stream, the second group describes some of the waste stream physical and chemical properties, and the third group specifies other properties of the waste stream including the number of years the waste stream is backlogged and the region in which it is generated.

Finally, the third type of indices, which are incidentally waste class specific, refer to the particular disposal technologies used at a given disposal site. They specify all the parameters required to perform impact calculations for the particular set of disposal technologies considered. These three types of indices are discussed below.

2.2.1 General Facility and Schedule Indices

There are two groups of indices in this category. The first group is called the general facility indices, and the second group is called the schedule indices. These indices are discussed below.

2.2.1.1 General Facility Indices

This group of indices is applicable to all the waste streams disposed at a given disposal site; they refer to properties such as the disposal site environment which are independent of the specific waste streams or the disposal technologies used.

There are five indices in this group: the region of the country in which the disposal facility is located (IR), the option on whether certain scenarios called the overflow scenarios are to be considered (IOFL), the option on facility buffer zone thickness (IBUF), the option on whether a certain type of classification is to be considered for certain waste streams (NBRN), and whether credit for certain waste form properties will be assumed in the radiological impact calculations (NBES). Alternative values for these indices are summarized in Table 2-11.

TABLE 2-11 . General Facility Indices

Symbol	Property	Optional Values
IR	Region Index	1 = Northeast ; 2 = Southeast; 3 = Midwest ; 4 = Southwest.
IOFL	Overflow Index	0 = Do Not Consider Overflow Scenarios 1 = Consider Overflow Scenarios
IBUF	Buffer Zone Index	1 = Standard Buffer Zone 2 = Larger Buffer Zone
NBRN	Barnwell Index	0 = Do Not Use Special Classification Tests 1 = Use Special Classification Tests.
NBES	Scenario Index	0 = Do Not Take Credit In Impact Scenarios For Some of The Waste Form Properties 1 = Take Credit For All Waste Form Properties

Region Index - IR

This index is a modified version of the region index used in the original Part 61 analysis methodology. The main rationale for this index is still to assign specific parameter values to the environmental characteristics affecting the calculation of short- and long-term impacts.

In this work, four reference site environments are characterized. These site environments are termed the northeast site, the southeast site, the midwest site, and the soutwest site, and are assigned IR index values of 1, 2, 3, and 4, respectively. A qualitative description of these four reference site environments has been listed in Table 2-4.

The parameters used in this report which are functions of site environmental properties are listed in Table 2-9. Further details of these environment specific parameters are presented in Appendix C.

Overflow Index - IOFL

This index provides the code user with an option to calculate impacts from a potential situation in which the disposal cells are allowed to accumulate leachate. An index value of 0 signifies that overflow scenarios will not be considered, and a value of 1 signifies that they will be.

If the option is implemented, radiological impacts from three overflow scenarios are calculated. The first occurs during disposal facility operation (leachate removal and treatment) while the last two occur after the closure of the facility (overflow of the disposal cells to a nearby stream, and treatment/evaporation of leachate withdrawn from the disposal cells). Overflow scenarios could occur in a humid site environment assuming a disposal technology equipped with an effective impermeable liner or at a site having soils with very low permeabilities. Some judgement, however, is required in the use of the option by the code user. It would obviously be inappropriate in an arid climate.

Buffer Zone Index - IBUF

This index affects the calculation of land acquisition costs and off-site environmental impacts. The land is purchased during the preoperational period, and the area purchased depends upon the amount of land needed for a given disposal capacity plus other facility requirements (e.g., administration area) including a buffer zone. This buffer zone varies depending on the option of the code user. The most conservative impact calculations result from specifying a buffer zone of 100 ft (IBUF value of 1). However, a buffer zone of 1000 ft may also be optionally used (IBUF value of 2). The manner in which this index is used is detailed in Appendix C.

Barnwell Index - NBRN

This index denotes whether a special type of classification test is to be used during the execution of the CLASIFY code (an index value of 1), or not (an index value of 0). This test is applicable only to certain candidate Class A waste streams and consists of a slightly different classification test to determine whether a waste stream should be stabilized in accordance with 10 CFR Part 61.56 (Ref. 3). In this test, the Part 61 Class A concentration limits are ignored and a waste stream is required to be stabilized if it contains any radionuclides having half lives exceeding 5 years and in concentrations exceeding 1 Ci/m^3. Details of this classification test are provided in Volume 2 of this report.

Scenario Index - NBES

This index denotes whether or not credit will be assumed for some of the waste form properties in calculation of impacts for some of the release scenarios. This index affects two specific waste form properties, namely dispersibility and leachability. These two properties are denoted through the waste form behavior indices I5 and I6, respectively (see Section 2.2.2.2). An NBES value of 0 denotes that the values of these two indices will be the least favorable values for impact calculation purposes, while an NBES value of 1 indicates that the values of these indices will be utilized as outlined in the impact scenarios.

2.2.1.2 Schedule Indices

This second group of indices contains five indices which relate to the schedule of disposal facility operations. They specify when the facility begins operation (IBEG), when the last waste is accepted (IEND), and the durations of the closure period (ICLS), post-closure observation period (IOBS), and the active institutional control period (IINS). These indices are summarized in Table 2-12.

TABLE 2-12 . Schedule Indices

Symbol	Property	Optional Values
IBEG	First Year of Operations	Greater than 1980
IEND	Last Year of Operations	Less than or equal to 2030
ICLS	Closure Period	A few years
IOBS	Observation Period	Five to ten years
IINS	Active Institutional Control Period	About 100 years

The First and Last Year of Operation - IBEG and IEND

The meanings of these indices are clear. IBEG denotes the first full or partial year of operation for the disposal facility, and IEND denotes the last full or partial year of operation for the disposal facility. However, it should be pointed out that a period of five years is implicitly assumed in the codes for pre-operational activities such as site selection, characterization, licensing, and construction. This five year period preceeds the year denoted by IBEG. The codes would have to be altered if this period significantly exceeds the assumed five years.

One related concept to the IBEG index is the number of years prior to the operation of the disposal facility that a particular waste stream(s) is stored -- i.e., the waste stream volume backlog which may exist at the time the facility is operational. This concept, however, must necessarily be waste stream specific since some of the waste streams can be stored more easily than others and some waste streams may be considered differently than others. For example, institutional waste streams may be potentially transferred across compact boundaries for disposal, while fuel cycle waste streams may have to be stored until a disposal facility is open within their compact region. This backlog volume is handled through the index IBLG (see Section 2.2.2.3).

Closure Period - ICLS

This index gives the duration of the post-operational closure period in years. Upon termination of disposal operations, the site operator is expected to bring the site to a passive maintenance mode through implementation of closure activities planned at the time of licensing. This period is expected to last one or possibly a few years.

Observation Period - IOBS

This index gives the duration of the post-closure observation period in years. At the end of the observation period, the operator is expected to apply for termination of the facility license. The licensing agency then will make the decision whether to terminate the license. This period is expected to last about five to ten years.

Institutional Control Period - IINS

This index gives the duration of the active institutional control period. The time period between the end of the observation period and the assumed breakdown in active institutional controls is denoted by IINS. This period is expected to last about 100 years.

2.2.2 Waste Stream Specific Indices

There are three major groups of waste stream specific indices. The first group, which was referred to as the waste processing indices in the original Part 61 analysis methodology, is primarily used to calculate impacts related to waste transportation and waste processing. The second group, which was referred to as the waste form behavior indices in the original Part 61 analysis methodology, indicates a certain property of the waste form or a specific calculational procedure to be used in the impact calculations. The third group of indices relate to other waste stream specific properties including the number of years the waste stream is backlogged prior to the operation of the disposal facility.

The first two groups of waste stream specific indices are stored in the codes in an array called ISPC. Consequently they are sometimes referred to as the ISPC indices.

2.2.2.1 Waste Processing Indices

These indices are waste stream specific and are primarily used to calculate waste transportation and processing impacts and costs. There are three waste processing indices: the waste processing parameters index, denoted by I1; the waste volume reduction factor, denoted by I2; and the waste volume increase factor, denoted by I3. These are briefly discussed below, and are detailed in Chapter 3.

Waste Processing Parameters Index - I1

Processing impacts include occupational exposures, population exposures, and costs. Population impacts from processing depend primarily on the radioactive contents of the waste streams and secondarily on the location at which the processing takes place. Only incineration (pathological incinerators and incinerator/calciners) is assumed to result in significant population exposures. Occupational exposures depend on the radiation environment in which the waste processing takes place as well as the waste activity. The costs of waste processing also depend on the size of the facility as well as the specific process being utilized. In addition, the packaging characteristics of a waste stream are specified in order to calculate exposures associated with waste loading, transportation, and emplacement at the disposal facility.

In order to account for these variations, six indices have been assigned to each waste stream in each waste spectrum and are utilized in the calculation of impacts. (Only five waste processing indices were used in the original Part 61 analysis methodology.) These six indices have been consolidated into a single index called the waste processing parameters index, I1, which combines the functions of the I1 and the I10 indices used in the original Part 61 analysis methodology.

The new I1 index is composed of the following six components: the waste packaging index, denoted by IPK; the waste processing index, denoted by IPR; the solidification index, denoted by ISL; the processing location, denoted by ILC; the processing environment, denoted by IEN; and finally, the processing radiation environment, denoted by IRE. The values assigned to these indices are summarized in Table 2-13.

The first index in Table 2-13, IPK, is unique in that it can be more than a single digit. It denotes the packaging characteristics of the waste stream as it is shipped. It is used to retrieve a set of five numbers that denote for a given waste stream what percentage is shipped in each of five modes: in large boxes, in small boxes, in drums, in small liners, and in large liners. It is waste stream specific, and the set of five numbers are used to calculate exposures and costs associated with waste transportation and disposal operations.

Volume Reduction Factor Index - I2

The Volume Reduction Factor (VRF) index and the following Volume Increase Factor (VIF) Index are used to express the physical effects of the various

TABLE 2-13 . Waste Processing Parameters Indices

Symbol	Property	Optional Values
IPK	Packaging Index	Record number of file containing distribution
IPR	Processing Index	0 = No volume reduction 1 = Regular compaction 2 = Improved compaction 3 = Hydraulic press 4 = Evaporation 5 = Pathological incineration 6 = Small calciner 7 = Large calciner
ISL	Solidification Index	0 = Unsolidified waste form 1 = Solidification scenario A 2 = Solidification scenario B 3 = Solidification scenario C 4 = High integrity container (HIC) packaging 5 = Stabilization through another means
ILC	Processing Location	0 = No processing 1 = Processing at the generator 2 = Processing at the disposal facility
IEN	Processing Environment	0 = No incineration 1 = Urban environment 2 = Rural environment
IRE	Processing Radiation Environment	1 = High facility background radiation environment 2 = Low facility background environment radiation

alternative waste processing techniques on a given waste stream. All waste processing techniques can be classified into one of two categories: (1) treatment technologies, such as compaction, evaporation, and incineration, in which there is one input and usually two output effluent waste streams, and (2) conditioning technologies, such as solidification and the use of absorbents, in which there is more than one input, and usually only one output waste stream (see VIF Index below).

The VRF is defined in this report as the ratio of the waste volume that is input to a given waste processing technology (untreated volume) to that of the treated waste volume (concentrated output volume). The index I2 is the VRF multiplied by 100. This is the same definition as the one used in the original Part 61 analysis methodology.

Volume Increase Factor Index - I3

Conditioning technologies usually lead to an increase in waste volume. This is due to the fact that nonradioactive physical substances such as solidification agents and/or absorbents are added to the waste stream and result in an overall increase of the waste volume.

The VIF is defined in this report as the ratio of the waste volume that is input to a conditioning technology (untreated volume) to that of the treated waste volume. The index I3 is the VIF multiplied by 100. This is the same definition as the one used in the original Part 61 analysis methodology.

2.2.2.2 Waste Form Behavior Indices

This group contains seven waste stream specific indices, five of which are similar to the six waste form behavior indices used in the original Part 61 analysis methodology. However, they have been modified in terms of the specific alternatives considered. The indices assigned to each waste stream for each waste spectrum (i.e., waste processing alternative) are as follows: an accident index denoted by I4; a dispersibility index denoted by I5; a leachability index denoted by I6; a chemical content index denoted by I7; a structural stability index denoted by I8; an activated metal index denoted by I9; and a source index denoted by I10. These indices are summarized in Table 2-14, and are discussed below.

Accident Index - I4

This index consists of two single-digit subindices which relate to the potential for waste forms to disperse into the atmosphere as a result of two types of hypothetical site operational accidents. The first subindex (ISC) is termed the scatter index while the second subindex (IFL) is termed the flammability index. These indices are assigned to allow the user to acquire some insights into the comparative levels of impacts mitigated through application of different requirements on waste form and packaging. The indices (and impact algorithms) should not be used as a substitute for specific test data on specific waste forms and containers.

TABLE 2-14 . Waste Form Behavior Indices

Symbol	Property	Optional Values
I4	Accidents First Digit: ISC Scatter	3 = Near zero 2 = Slight to moderate 1 = Moderate 0 = Severe
	Second Digit: IFL Flammability	3 = Non-flammable 2 = Low flammability 1 = Burns if heat supplied 0 = Flammable
I5	Dispersibility	3 = Near zero 2 = Slight to moderate 1 = Moderate 0 = Severe
I6	Leachability Index	1 = Unsolidified waste form 2 = Solidification scenario A 3 = Solidification scenario B 4 = Solidification scenario C
I7	Chemical Content	0 = No chelating agents or organic chemicals 1 = Chelating agents or organic chemicals likely present
I8	Stability	0 = Structurally unstable waste 1 = Solidified waste 2 = Structurally stable solidified waste 3 = Stabilized using high-integrity containers 4 = Stabilized by other means
I9	Activated Metal	0 = Non-activated metal waste 0 < Activated metal waste
I10	Sources	0 = Non-source waste 0 < Source waste

The scatter index (ISC) is a measure of the potential for the contents of a waste container to be dispersed into the atmosphere due to an operational accident in which the waste container is severely damaged. This could hypothetically occur from an accident in which the waste container is dropped from a significant height. Waste forms which are assumed to have a low probability of becoming suspended into respirable particles are assigned an ISC index of 3. Waste forms which are assumed to have a high probability of becoming suspended into the atmosphere (such as incinerator ash) are assigned an ISC index of 0. Waste forms which tend to crumble or fracture extensively are assigned an index value of 1. Waste forms consisting of a mixture of material with ISC indices of 1 and 3 are assigned an index value of 2.

The second subindex (IFL) ranks waste forms according to their flammability. Waste forms which are judged to not burn even on prolonged exposure to open flame and moderately intense heat are assigned an index of 3. These consist of waste forms that should experience no evidence of combustion or decomposition upon exposure to 1000 F for 10 minutes. Those waste forms that will sustain combustion are assigned an index of 0. These include waste forms such as liquids with flame points around 600 F. Between these extremes are two additional flammability categories. Waste forms which show evidence of combustion and/or decomposition upon exposure to 1000 F for 10 minutes but will not sustain burning when the heat source is removed are assigned an index of 1. Waste forms consisting of a mixture of materials with flammability indices 1 and 3 are assigned an index value of 2. The specific manner in which these indices are used is discussed in Sections 4.1.3 and 4.6.

Dispersibility Index - I5

This index is a measure of the potential for suspension of radioactivity should the waste form be exposed to wind or mechanical abrasion after a significant period (on the order of 100 years). That is, this index is a measure of the degree to which individual waste streams may be suspended as respirable particles into the air by wind or the action of a potential inadvertant intruder. The authors recognize that it is very difficult, if not impossible, to predict the actual dispersibility of waste forms in the future. This index is therefore considered to enable the code user to illustrate the usefulness of nondispersible waste forms in mitigating long-term impacts.

Waste forms which are assumed to have a high probability of becoming suspended into respirable particles are assigned an index value of 0. Those waste forms which are assumed to have a low potential of becoming suspended are assigned an index value of 3. Waste forms which tend to crumble or fracture extensively and those forms that are subject to relatively rapid (within about 100 years) decomposition are assigned an index value of 1. Waste forms consisting of a mixture of materials with dispersibility index values of 1 and 3 are assigned an index value of 2.

Given the existing uncertainty concerning the appropriateness of this index, the user is given an option to cancel its effect through the use of the scenario index NBES (see Section 2.2.1.1). That is, whenever NBES has a value of 0, an I5 index value of 0 is assumed.

Leachability Index - I6

This index is a measure of a waste form's resistance to leaching and is primarily determined by the solidification procedures used. Unsolidified waste forms, which are assumed to be readily leached, are asssigned an index of 1. Waste streams solidified according to solidification scenarios A, B, and C are assigned indices of 2, 3, and 4, respectively.

The solidification scenarios represent idealistic assumptions regarding varying levels of performance that can be achieved through available solidification techniques. In this report, a level of performance designated by solidification scenario A has been simulated by assuming that the waste is solidified using cement; a level of performance designated by solidification scenario B has been simulated by assuming that the waste is solidified using synthetic organic polymers; and a level of performance designated by solidification scenario C has been simulated by assuming that all of the waste is solidified using optimistically improved synthetic organic polymers.

This index is used in a similar manner as that in reference 8. The authors believe that insufficient information is currently available to assign definite radionuclide-specific leaching parameters to solidified waste forms under actual disposal site conditions. Rather, the leachability index is included to allow the code user to assess what level of overall impact reduction can be achieved given varying assumptions on solidified waste form leachability. This is significant since only a portion of the waste delivered to a disposal facility is in a solidified form, and despite improvements in solidified waste form leachability, eventually a situation will result in which releases from other waste forms will overshadow those from solidified waste.

This index may be used in the leachate accumulation, groundwater and intruder-agriculture scenarios. In the leachate accumulation and groundwater scenarios, I6 affects the levels of long-term radionuclide release, and in the intruder-agriculture scenario, it affects the food (soil) uptake pathway since the level of contamination in interstitial soil water available to vegetation may depend on the leachability of the waste. The use of the leachability index is further discussed in Section 4.1.3. The values assigned to the index, I6, however, may be modified further depending on properties of the waste and the disposal technology implemented (see Section 4.1.3).

In a manner similar to the I5 index, the user is given an option to cancel the effect of this index through the use of the scenario index NBES (see Section 2.2.1.1). That is, whenever NBES has a value of 0, an I6 index value of 1 is assumed.

Chemical Content Index - I7

This index denotes whether a waste stream may contain chelating or organic chemicals that may increase the mobility of radionuclides during and/or after leaching. An index value of 0 indicates the likelihood that these agents are absent in the stream, whereas an index value of 1 indicates that the stream is likely to contain chelating or organic chemicals. Its use, in conjunction with other indices including I6 and IS, is discussed in Section 4.1.3.

Stability Index - I8

This index primarily denotes whether the waste form is likely to reduce in volume after disposal due to compressibility, large internal void volume, and/or chemical and biological attack. A secondary function of the index is to specify the stabilization technique used during waste processing.

Briefly, this index denotes whether the waste is structurally unstable (an index value of 0), whether it has been solidified (an index value of 1 implies solidification, and an index value of 2 denotes that the waste is structurally stable and solidified), whether it has been stabilized through the use of a high integrity container (an index value of 3), or whether it has been stabilized by other methods (an index value of 4). In general, the value of I8 has been assigned based on the physical descriptions of the waste provided in Appendix A of this report and the processing alternative being considered (see Appendix B). For example, waste streams with a large void volume or potential for compressibility/ biodegradability (e.g., trash) have been assigned an index value of 0. A waste stream which can be considered to be stable without the need of processing (e.g., a large stainless steel component) is assigned an index value of 4.

It should be noted that the solidification of a waste stream does not necessarily imply that the waste is structurally stable. Only waste streams solidified using methods consistent with the requirements of the Low-Level Waste Licensing Branch Technical Position on Waste Form (Ref. 7) are assumed to be structurally stable in accordance with the requirements of 10 CFR Part 61.56. Solidification of these waste streams must be conducted in accordance with a process control program that ensures stability of the waste when subjected to compression, immersion, and biological attack. A leaching standard must also be met. These waste streams are assigned an I8 index value of 2. Other solidified wastes, which only meet a standard that the waste product be in a free standing form, are assigned an I8 index value of 1. The specific use of this index is discussed further in Section 4.1.3.

Activated Metal Index - I9

This index signifies whether the waste stream is an activated metal waste stream. It triggers the use of correction factors for activated metal waste streams for which the distribution of radionuclides within the waste is such that the radionuclides are not as easily accessible as other waste streams to transfer agents such as wind or water. This index was called the accessibility index in the original Part 61 analysis methodology, and denoted a similar property of the waste streams; however, its use has been slightly expanded as detailed below.

Most waste streams contain surface contaminated materials or materials containing radioactivity in readily soluble forms; these waste streams are assigned an activated metal index of 0. This value indicates that the waste form affords no additional protection in impact calculations. Most waste streams, including solidified streams, fall into this category.

The waste streams that are almost exclusively activated metals with imbedded radioactivity not readily accessible to the transfer elements are assigned an index of 1 or higher. The specific value of this index is used to access information (from a file called METALS.DAT) on its interaction with the environment. These properties are as follows: (1) the time period (in years) it will take for the waste stream to corrode completely, (2) the air dispersibility of the corrosion products, (3) the water solubility of the corrosion products, and (4) the self shielding afforded by the waste stream against direct radiation. These four properties are discussed in Section 4.1.3.

Source Index - I10

This index has been formulated for this report and denotes whether the waste stream is a source waste stream (i.e., those waste streams with high activities in very small volumes). Sources may be considered in a very different manner than other waste streams in this report. This difference is primarily due to the option that allows the total activity per source, instead of activity per unit volume, to be considered during waste classification and impact calculations. This index signifies that option, and also permits access to certain correction factors for impact calculations in a manner similar to the activated metal index, I9.

All waste streams for which radiological properties are specified in concentrations (i.e., activity per unit volume) are assigned a source index value of 0. Most waste streams fall into this category. Any other positive index denotes a source waste stream for which the radiological properties are specified in units of total activity per source. The specific value of this index is used to access information (from a file called SOURCE.DAT) on its interaction with the environment. These properties are as follows: (1) the time period after closure (in years) during which the source can be assumed to be inaccessible to the transfer agents, (2) the dispersibility of the source, (3) the water solubility of the source, and (4) the self shielding afforded by the waste stream against direct radiation. These properties are discussed in Section 4.1.3.

2.2.2.3 Other Waste Stream Specific Indices

These indices, which are summarized in Table 2-15, denote other waste stream specific properties. There are two groups of other waste stream specific indices: waste volume related, and waste classification related.

TABLE 2-15 . Other Waste Stream Specific Indices

Symbol	Property	Optional Values
Name	Waste Stream	Given in Table 2-6
IRI	Generating Region	1 = Northeast 2 = Southeast 3 = Midwest 4 = Southwest
IBLG	Years of backlog	Any value as long as IBEG-IBLG > 1980
FVOLI	Volume Fraction	Fraction of the waste volume generated in region IRI and shipped to facility in region IR
NDXS	Classification Index	0 = Do not consider waste stream 1 = Normal waste stream 2 = Normal waste stream, can be stabilized only through disposal technology -1 = Special classification test can be applied -2 = Special classification test can be applied, and the waste stream can be stabilized only through disposal technology
NDST	Distribution Index	0 = Waste does not have a distribution >0 = Waste has a concentration or activity distribution

There are three indices and a parameter in the first group: the first index is represented by the waste stream name given in Table 2-6, and is used to access projected volumes of the waste stream by regions; the second index is denoted by IRI, and signifies the region in which the waste is generated; and the third index is denoted by IBLG, and signifies the number of years this waste stream is stored pending the operation of the particular disposal facility (i.e., years of backlog). The parameter is denoted by FVOLI, and signifies the fraction of that waste stream that is shipped from region IRI to the disposal facility. These three indices and the parameter are read from the file INPUTS.DAT.

The second group contains two indices: the first index is denoted by NDXS, and signifies whether the waste stream is to be considered in a particular computer run, the type of classification, and its amenability to being stabilized through disposal technology rather than processing or packaging by the generator. The second index is denoted by NDST, and signifies whether the total concentration or activity of the waste stream has been observed to have a range of values depending on the generator and waste management practices (i.e., whether the waste stream concentration or activity is distributed). Both of these indices are used by the code CLASIFY, and are read from the file called WASCAR.DAT. The alternative values that can be assigned to these indices are presented in Table 2-15.

Waste Stream Names and Projected Volumes

In this report, waste stream volumes that are projected to be generated are handled through a separate minor code called VOLUMES. There are several reasons for this separation. First of all, volume projections are involved and complicated; handling them separately reduces the complexity of the major codes. Second, volume projections are needed only by the IMPACTS, INTRUDE, and ECONOMY codes; the CLASIFY and INVERSE codes do not require them for their execution. Finally, waste volume projections are constantly changing due to a number of reasons including delay or cancellation of nuclear power reactor construction, implementation of more efficient volume reduction technologies, and potential implementation of decisions such as reprocessing of spent fuel. Thus, a major portion of the uncertainty is compartmentalized by handling volume projections through a separate code.

The code VOLUMES is summarized in Section 2.4 and detailed in the second volume of this report. It is based on the data and information presented in Appendix A, and it creates a file called VOLUME.DAT that contains the projected volumes for each waste stream given in Table 2-6 for each of the four regions considered in this study, and for each of the years between 1981 through 2030, inclusive. This file is accessed by the code IMPACTS through the use of waste stream name, region in which it is generated, IRI (see below), and the year of interest.

Generating Region Index - IRI

This index is similar to the region index IR but signifies the region in which the waste stream is generated (which may be different from the region in which the waste stream is disposed). Integer values for this index range from 1 to 4 and denote the same regions as the IR index. This index is also used to analyze transportation impacts.

Backlog Index - IBLG

This index gives the number of years that the specific waste stream has been stored pending the establishment of the disposal facility. It is used, in conjunction with the schedule index IBEG, to modify the waste volumes assumed to be disposed during the first year's operation at the disposal facility.

Waste Stream Type - NDXS

This index is used by the code CLASIFY and read from the file WASCAR.DAT. The values, however, may be modified through input in CLACON.DAT during the execution of CLASIFY. A value of 0 for this index signifies that the waste stream is to be excluded from the analysis, which allows the user to consider the relative significance on costs and impacts of any given waste stream or group of waste streams. Positive values signify that routine classification tests given in 10 CFR Part 61 are to be used for this waste stream; negative values denote waste streams which may be subject to the optional classification test (NBRN=1) discussed in Section 2.2.1.1. In addition, an absolute value of 1 (normal case) implies that the waste stream is stabilized, if need be based on classification tests, independent of the disposal technology -- i.e., by processing or packaging by the generator. An absolute value of 2 implies that the waste stream can only be stabilized by use of a disposal technology that provides stability.

(The index NDXS used in the original Part 61 analysis methodology signified both the distributional nature of the waste stream (this property has been assigned to the NDST index discussed below) and part of the current use of this index. It is now used to indicate whether the waste stream will be subjected to the special classification tests.)

Waste Stream Distribution Index - NDST

It was recognized in the original Part 61 analysis methodology that in practice there exists a wide variation in the total concentrations of almost all of the waste streams shipped for disposal. This variation results from many factors including different waste management practices by the waste generator, different ages and designs of waste generating facilities, and so forth. The final EIS considered seven waste streams for which a gross concentration volume distribution could be defined. For each waste stream, a series of concentration ranges were defined, and for each concentration range the fraction of the waste volume that fell within the range was determined as well as the average concentration across the range. The details of the methodology used to present and consider the distributions in the calculations are presented in Appendix C of the final EIS on Part 61 (Ref. 2).

This report also considers potential distributions of waste streams. The index NDST performs this function. An index value of 0 signifies that the waste stream does not have a distribution, while a value greater than 0 specifies that there is a distribution.

2.2.3 Disposal Technology Indices

This group of indices are disposal technology (DT) specific and correspond to some of the DT indices used in the original Part 61 analysis methodology. Several of the original DT indices (see Table 2-5), such as IR and IINS, have been considered in the Section 2.2.1. The remaining indices, which are summarized in Table 2-16, have been expanded considerably to enable more accurate consideration of a much larger set of possible disposal technologies.

TABLE 2-16 . Disposal Technology Indices

Symbol	Property	Optional Values
IU	Utilization Index	1 through 6. Gives the number of the lowest class of waste that the waste is disposed with.
ID	Disposal Technology	Record number of file containing the properties EFF,SEF,DPT,DTK
IT	Topmost Waste	1 through 6. Gives the number of the topmost waste class above the waste class considered.
IC	Cover Index	0 = Waste is disposed underneath another class of waste 1 = Regular cover 2 = Improved cover
IE	Emplacement Index	1 = Random disposal 2 = Stacked disposal
IB	Backfill Index	1 = Natural soils 2 = Imported sand or gravel 3 = Grout
IX	Compaction Index	1 = Regular 2 = Improved 3 = Extreme
IS	Chemical Segregation Index	0 = No segregation 1 = Segregation of waste with chelating agents and/or chemicals

An important concept associated with these DT indices is the disposal technology configuration which must be specified for each computer run. The user must specify all the DT indices for each disposal technology configuration and assure that they are compatible. The descriptions of the DT indices are presented in Section 2.2.3.1, and a representative example disposal technology configuration is discussed in Section 2.2.3.2.

2.2.3.1 Descriptions of the Disposal Technology Indices

All the DT indices are six dimensioned arrays -- i.e., there are six values for each index. This is because that there may be as many as six, and as few as one, different disposal technologies used simultaneously at a given disposal site. These six disposal technologies and the corresponding waste classes considered in this report are as follows:

1 - Class A,
2 - Class A Stable,
3 - Class B,
4 - Class C,
5 - Class D1, and
6 - Class D2.

All these wastes except 1 - Class A are assumed to be structurally stable in accordance with the provisions of 10 CFR 61.56(b) and the NRC Low-Level Waste Licensing Branch Technical Position on Waste Form (Refs. 3 and 7).

Utilization Index - IU

This index is also referred to as the geometry and configuration index. It gives the lowest order of waste class that the specific waste class is mixed with during disposal. This index incorporates a portion of the function of IS index of the original Part 61 analysis methodology (see IS index below). For example, if IU(3)=1, then Class B waste will be disposed together with Class A waste. If IU(3)=2, then Class B waste will be disposed together with Stable Class A waste. The value of this index cannot be greater than the index, i.e., IU(I) must be less than or equal to I. This index is used in impact calculations as well as cost calculations.

As stated previously, segregation of structurally stable and structurally unstable waste classes is required by Part 61 unless the unstable wastes are stabilized in place through other methods, e.g., concrete vaults. Such segregation increases the performance capability of the disposal cell covers by limiting expected long-term disposal cell subsidence to those disposal cells containing only the structurally unstable wastes.

Disposal Technology Index - ID

This index denotes the disposal technology to be utilized for a given class of waste. It conceptually replaces the ID index of the original Part 61 analysis methodology, and triggers specific numerical values for the various disposal technology parameters presented in Table 2-16. The index ID gives the record number of the file (called DISTEC.DAT) storing these parameters plus other information on the disposal technology, such as costs. In this report, each different disposal technology is assigned the four parameters specified in Table 2-16 and discussed below.

EFF - Volumetric Disposal Efficiency

This parameter is conceptually identical to that used in the original Part 61 analysis methodology. It is given as the the volume of disposal space avalable in a disposal cell (in m^3) divided by the surface area (in m^2) of the disposal cell. The surface area of the disposal cell is defined to be the horizontal plane that is in contact with the waste and closest to the ground surface.

SEF - Surface Disposal Efficiency

This dimensionless parameter is also identical to the concept used in the original Part 61 analysis methodology. It denotes the ratio of the surface area occupied by disposal cells (surface area defined as in EFF above) to the surface area of the disposal cells plus the surface area around and between the disposal cells that have not been utilized for disposal. Subjective judgement may be required to define the area around the disposal cells which is not available for disposal.

DPT - Disposal Cell Depth

This is a parameter formulated for this report. It denotes the depth of the top of the disposal cell below the top of the final disposal cell cover (in meters).

DTK - Disposal Cell Thickness

This parameter has also been formulated for this report. It gives the maximum height (in meters) to which waste can be placed in the disposal cell. For waste classes that are disposed in a segregated manner from other waste classes, the value specified in the file DISTEC.DAT is used. However, for waste classes that are disposed mixed with other waste classes, it is calculated internally using actual volumes disposed.

These four parameters, together with the other indices discussed below, describe a specific disposal technology in sufficient detail to enable the performance of the impacts assessments discussed later in this report.

Topmost Waste Index - IT

This index denotes the lowest waste class that is immediately below the final disposal cover (i.e., the topmost waste class) and above the waste class in question. It is predicated on the possibility that some wastes may be disposed in a separate disposal technology below other wastes. An example might be disposal of Class D wastes in auger holes in a trench that might contain Class C wastes at the trench bottom and Class B and Stable Class A wastes at the top. (IT=2 for this example.) This index is also used in the groundwater scenarios as well as cost calculations.

Waste Cover Index - IC

This index is similar to the IC index of the original Part 61 analysis methodology, and is used in intruder and groundwater calculations as well as cost calculations. It is set equal to zero for waste classes for which the IT index does not correspond to the waste class.

There are two cover options in this report. An index value of 1 generally denotes a reference cover composed of 1 m of soil below grade and 1 m of soil above grade. An index value of 2 generally denotes an improved cover composed of 1 m of compacted soil below grade and 1 m of engineered cover (including low-permeability layers and topgrass for humid climates and topgravel for arid climates) above grade.

Emplacement Index - IE

This index denotes the specific method used to emplace the waste in the disposal cells. Associated with each index value is an emplacement efficiency (EMP) which is defined as the volume of wastes emplaced in the disposal cell per unit volume of available disposal space. The options considered in this report are discussed below.

> Random emplacement (index value 1) involves simply dumping the waste directly into the disposal cell. It is the fastest waste emplacement method which can be used, and therefore leads to the lowest occupational exposures at the disposal facility. However, random emplacement of waste containers may be accomplished with only about 50% emplacement efficiency (one-half of the available space is empty or filled with earth or other material), and there is a higher probability of the occurrence of accidents as well as container damage.
>
> Stacked emplacement (index value 2) involves stacking waste containers in neat piles, using cranes, fork lifts, etc. This technique represents the maximum practical volume utilization. In this case, the potential for accidents and waste container damage is much lower, and approximately 75% of the available disposal space is used - i.e., the emplacement efficiency is 0.75. However, additional fuel must be used to operate the heavy equipment used for emplacement, and occupational exposures increase as more men must spend more time near the emplaced waste.

Backfill Index - IB

This index denotes the type of backfill used to fill the interstitial spaces between the waste packages. An index value of 1 correspond to the use of natural excavated soils as backfill material, an index value of 2 corresponds to the use of imported sand and/or gravel as backfill material, and an index value of 3 corresponds to the use of grout that sets in place and improves the disposal cell stability.

In the past, natural soils dug during the excavation of the waste trenches were almost exclusively used as backfill material. Several operational problems were experienced as a result of this practice. For example, uneven compaction of natural soils sometimes resulted in formation of dams within the trenches which hydraulically isolated sections of the disposal cell from each other. Another common problem was that natural soils frequently did not flow to fill all the interstitial spaces; this resulted in void spaces within the trenches that subsequently collapsed and led to trench cover failures.

An option is therefore included whereby an improved backfill (sand and/or gravel) is used that expedites filling the voids between the waste packages. As in the original Part 61 analysis methodology, this option is assumed to involve additional disposal costs.

The third option involves the use of grout to fill the spaces between the waste packages. During the grouting operation, as each layer of waste is emplaced in the disposal cell, pumpable concrete (grout) is pumped to fill all the interstitial spaces. Grouting is expensive, but its use is advantageous in that the waste is totally encapsulated and immobilized. There is little opportunity for infiltrating precipitation to contact the waste, the grout provides stability, and potential long-term migration and intruder impacts are minimized. This option was available in the original Part 61 analysis methodology through the use of the old IG index.

Compaction Index - IX

The value of this index can be either 1, 2, or 3, and it signifies the extent to which the disposed wastes and backfill are compacted. Such compaction measures, which are especially significant for non-engineered disposal cells containing unstable wastes, are assumed to be implemented during disposal operations.

For emplaced wastes and backfill, a compaction program involving no special compaction other than that incidentally provided by the weight of heavy equipment is denoted by an index value of 1. A more extensive compaction program is denoted by an index value of 2. This could involve use of specialized equipment such as sheeps-foot rollers or vibratory compactors. Finally, a compaction program involving very extensive techniques such as dynamic compaction or similar measures is assigned a value of 3.

Chemical Segregation Index - IS

This index is a conceptually reduced version of the IS index of the original Part 61 analysis methodology. It now denotes segregation of waste potentially containing chelating agents or organic chemicals from waste that does not contain these chelating agents and chemicals. Its value is either 0 (denoting no segregation) or 1 (denoting segregation).

2.2.3.2 Example Disposal Technology Configuration

This section considers a hypothetical disposal facility and gives examples of the DT indices which would correspond to this facility. The following is a description of the disposal facility, which is assumed to be located in an arid environment.

- Unstable Class A waste is segregated from other wastes and is disposed in a random emplacement mode in a shallow trench. This trench is backfilled with natural soil, extreme compaction measures are used, and a regular cover is installed.

- Stable Class A waste is disposed in a separate deep disposal trench together with Classes B, C, and D1 wastes in a random emplacement mode. This trench is backfilled with imported sand, regular compaction is used, and an improved cover is installed. Class C waste is disposed so that at least 5 m of soil or lower activity waste separates the waste from the earth's surface, while Class D1 waste is disposed so that at least 10 m of soil or lower activity waste separates the waste from the earth's surface.

- Class D2 waste is disposed in a stacked emplacement mode in separate concrete engineered bunkers which are grouted. A regular cover is installed.

In addition, segregation of wastes containing chelating agents and/or organic chemicals is implemented. The following Table 2-17 presents a summary of the DT indices for this configuration.

TABLE 2-17 . Example Disposal Technology Configuration

DT Index	Unstable Class A	Stable Class A	Class B	Class C	Class D1	Class D2
IU	1	2	2	2	2	6
ID	4	3	3	3	3	11
IT	1	2	2	2	2	6
IC	1	2	0	0	0	1
IE	1	1	1	1	1	2
IB	1	2	2	2	2	3
IX	3	1	1	1	1	1
IS	1	1	1	1	1	1

The meanings of the individual index values assigned can be located in Table 2-16. Most of the values in this table are self-explanatory except the ID index.

The disposal technologies and basic parameter values considered in this report are summarized in Table 2-18. As can be seen, a small shallow trench (10 m wide x 60 m long x 8 m deep) in an arid environment is designated as ID = 4, a large deep trench (30 m wide x 180 m long x 14 m deep) is designated as ID = 3, and the concrete trench in an arid environment is designated as ID = 11.

TABLE 2-18 . Disposal Technologies Considered and Parameters

Disposal Technology and ID Number	Location	Basis	Volumes (m^3)					
			Empty Cell	Maximum* Waste	EFF**	SEF	DPT**	DTK**
Regular								
1.Large Trench	Humid	each	38,326	24,370	6.23	.88	2.0	6.7
2.Small Trench	Humid	each	2,763	1,640	3.88	.69	2.0	4.7
3.Large Trench	Arid	each	65,539	45,140	11.4	.88	2.0	13.0
4.Small Trench	Arid	each	3,533	2,210	5.25	.69	2.0	7.0
5.Slit Trench	Humid	each	1,209	739	4.73	.47	2.0	4.7
6.Slit Trench	Arid	each	1,227	752	4.82	.47	2.0	5.0
7.Auger	Humid	40 grp	1,612	997	4.70	.079	2.0	4.7
8.Auger	Arid	40 grp	8,484	6,150	29.0	.079	2.0	29.0
Concrete								
9.Trench	Humid	each	5,033	4,900	5.7	.44	2.0	5.7
10.Trench	Arid	each	5,033	4,900	5.7	.44	2.0	5.7
11.Slit Trench	Humid	6 grp	1,316	937	5.7	.17	2.0	5.7
12.Slit Trench	Arid	6 grp	1,316	937	5.7	.17	2.0	5.7
13.Repackaging	Humid	each	28,690	12,500	6.75	.68	2.0	6.75
14.Repackaging	Arid	each	28,690	12,500	6.75	.68	2.0	6.75

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* Maximum waste volume is for stacked disposal; it may be recalculated. See Chapter 5.

** These values may also be recalculated; see Chapter 5.

2.3 Overview of Pathway Analysis Methodology

There are many diverse mechanisms through which radionuclides contained in the waste streams may be potentially released (i.e., mobilized from the waste and become accessible to a transport agent such as wind or water), transported through the environment (i.e., moved from one location to another through the atmosphere or soil by a transport agent), and thereby become accessible to humans through various pathways. Human access to the radioactivity may result either through direct human contact with contaminated material (e.g., inhalation of air, ingestion of water, or direct exposure to radiation) or indirectly through contaminated biota (through a multitude of pathways involving ingestion and exposure) which have come into contact with contaminated material. Each of these radionuclide release/transport/pathway combinations represent a complex series of interactions which are affected by a wide range of parameters such as waste properties, treatment or disposal facility environmental conditions, and operational procedures.

The basic procedure used in this report to describe these various radionuclide release/transport pathway combinations is that developed in References 1 through 4. The reader may consult these references for additional details. A summary, however, is provided below.

2.3.1 Generalized Approach

There are two general types of radiological calculations performed in this report. The first type is to calculate radiological impacts given a number of waste streams having specific radionuclide concentrations under specified disposal conditions, and the second type is to calculate the limiting concentrations or activities within a waste stream so that a specified dose rate will not be exceeded under specified disposal conditions. These calculations are referred to as impacts (or as forward) and as inverse (or as backward) calculations, respectively. These are summarized below.

Impacts Calculation

In order to perform the impacts calculation, first a given waste stream, denoted by subscript (i), is classified into one of six potential classes, henceforth denoted by subscript (j). This classification is done using the code CLASIFY (see Section 2.4). The volume (in m^3) of waste stream (i) classified in waste class (j) is denoted by V_{ij}. Thus, the total volume of waste V_j in m^3 disposed in waste class (j) becomes the sum of V_{ij} over (i), i.e.,

$$V_j = \sum_i V_{ij} \tag{2-1}$$

After classification, the next step in the forward calculation is the calculation of impacts resulting from the volume V_{ij}. This is performed in the code IMPACTS (see Section 2.4). For a given release scenario, IMPACTS uses the following general equation:

$$H_{ij} = \sum_{n} C_{ijn} \times I_{ijn} \times PDCF_n \qquad (2\text{-}2)$$

where H_{ij} is the dose rate to the individual in units of mrem/year resulting from waste stream (i) in waste class (j), where the impacts are summed over all individual radionuclides (n) in the waste, and where:

C_{ijn} = the concentration of radionuclide (n) in the (i)th waste stream in the (j)th waste class considered;
I_{ijn} = an interaction factor relating the concentration of the (n)th radionuclide in the (i)th waste stream in the (j)th waste class to the concentration of the radionuclide at the biota access location; and
$PDCF_n$ = the pathway dose conversion factor (PDCF) for that radionuclide, generally in units of mrem/year per Ci/m^3.

Radionuclide concentrations (C_{ijn}) are generally given in units of Ci/m^3. For some waste streams and impact scenarios, however, radionuclide activities (units of Ci) are used in equation (2-2) rather than concentrations. The factor (I_{ijn}) represents the movement of the radionuclide from the waste to a biota access location. It is generally dimensionless, but has units of m^{-3} when the waste stream activities rather than concentrations are used in equation (2-2). The factor $PDCF_n$ represents the resulting human exposures resulting from a unit concentration of the radionuclide at the biota access location, and assuming that these exposures are received continuously over a year. (Exposures lasting less than a year are handled by incorporating an appropriate factor into I_{ijn}.) A biota access location is defined as any location where a human may receive an exposure to radioactive isotopes due to inhalation, ingestion, immersion in a contaminated cloud, or proximity to a gamma-emitting source.

The concept of waste stream volume in m^3, V_{ij}, is clearly defined for routine and activated metal waste streams, and is directly used in the calculations. However, for source waste streams, the number of sources, N_{Sij}, is used in the calculations. Thus, for source waste streams, the following equations are applicable:

$$V_{ij} = N_{Sij} \times V_{Si} \qquad (2\text{-}3)$$

and

$$d_{Sij} = N_{Sij} / V_j \qquad (2\text{-}4)$$

where N_{Sij} is the number of sources in waste stream (i) in waste class (j), V_{Si} is the packaged volume per source for a source waste stream defined through the index I10 (see Section 2.2.2), and d_{Sij} is the density of the (i)th source waste stream within waste class (j).

The next step in the forward calculation is the calculation of impacts due to a given waste class, i.e., H_j. It clearly is not equal to the direct sum of H_{ij} over (i); the volumes of waste involved must be considered. It is given by the following equation:

$$H_j = \sum_i \sum_n (V_{ij} \times C_{ijn} / V_j) \times I_{ijn} \times PDCF_n \qquad (2\text{-}5)$$

or

$$H_j = \sum_i (V_{ij}/V_j) \times H_{ij} \qquad (2\text{-}6)$$

This equation takes the total activity in a waste stream (i) in waste class (j), i.e., $V_{ij} \times C_{ijn}$, and distributes it over the entire volume within that waste class, i.e., V_j.

The next step involves the calculation of total impacts, H_t, from the facility. There are two possible ways to do this. The first way involves volume averaging similar to equation (2-6), i.e.,

$$H_t = \sum_j (V_j/V_t) H_j \qquad (2\text{-}7)$$

where V_t is the sum of V_j over (j). This is the approach actually adopted in this report. The second way involves using disposal site areas, i.e.,

$$H_t = \sum_j (S_j/S_t) H_j \qquad (2\text{-}8)$$

where S_j is the total area (including the spaces between the disposal cells) attributable to (j)th waste class, and S_t is the sum of S_j over (j), i.e.,

$$S_j = V_j / (EMP_j \times EFF_j \times SEF_j) \qquad (2\text{-}9)$$

and

$$S_t = \sum_j S_j \qquad (2\text{-}10)$$

where EMP_j is the emplacement efficiency (dimensionless), EFF_j is the volumetric disposal efficiency (in m), and SEF_j is the surface disposal efficiency (dimensionless) of the (j)th waste class.

However, basic disposal technology parameters given in Table 2-18 must be recalculated under certain conditions. This subject is discussed further in Chapter 5.0.

The above formulation is repeated for each distinct release/transport/pathway scenario. Generalized equations similar to the above equations are applicable for exposures to populations. Further discussion on the general approach to calculating the release and transport of radionuclides through environmental pathways is provided in Section 2.3.2. The pathway dose conversion factors are discussed in Section 2.3.3.

Inverse Calculation

As stated, it is frequently desirable to calculate the limiting concentrations or activities within a waste stream so that a specified dose rate will not be exceeded under specified disposal conditions. To do this, some modifications are made to equation (2-2), the generalized approach to calculating potential radiological impacts to an individual. These modifications can be envisioned as forming the "inverse" of equation (2-2). For a given scenario, the generalized form of this inverse is as follows:

$$L_{jn} = DLC / (I_{jn} \times PDCF_n) \qquad (2\text{-}11)$$

In this equation, the parameters I_{jn} and $PDCF_n$ remain essentially as discussed above, the subscript (n) refers to the particular radionuclide considered, and the subscript (j) refers to the waste class. L_{jn} is the limiting radionuclide concentration (Ci/m^3) or activity (Ci) so that DLC is not exceeded according to a specific combination of disposal conditions (e.g., disposal method or waste class, waste form, decay time). Limiting radionuclide activities are calculated for source waste streams while limiting radionuclide concentrations are calculated for all other waste streams.

DLC stands for the impact scenario specific dose limitation criteria. It is in units of mrem/yr (total body equivalent) and is input by the user of the computer codes (See Volume 2).

2.3.2 Radionuclide Release and Transport

The interaction factor I (dropping the subscripts for simplicity) discussed in the previous section is composed of four subfactors (Ref. 6) as follows:

$$I = f_o \, f_d \, f_w \, f_s \qquad (2\text{-}12)$$

where:

f_o = time delay factor
f_d = site design factor
f_w = waste form and package factor
f_s = site selection factor

The factor (f_o) accounts for the radionuclide decay that would take place between the time the waste stream leaves the waste generator's hands and the time that contact is made by the transport agent; the factor (f_d) accounts for the inherent design characteristics of the waste disposal facility which influences the release and/or transport of radionuclides; the factor (f_w) accounts for the physical and chemical characteristics of the waste, at the time of the initiation of the release/transport scenario, that may inhibit radionuclide transfer by the transport agent; and the factor (f_s) includes the effects of the environment on radionuclide release/transport.

As a hypothetical example, consider the following scenario involving waste which may be assumed to be indistinguishable from dirt at some time following disposal. After the disposal facility ceases operations, there will be an active institutional control period. Following this period, the facility is assumed to be accessible to inadvertant intruders. One of these intruders decides to build a house on top of the portion of the facility containing the waste assumed in this example, and excavates a hole for his cellar (also called the Intruder-Construction Scenario - see Chapter 4). During this activity, a portion of the waste is dispersed into the air as particulates, combining with the dust raised during excavation operations. To be able to calculate the possible radiological exposures received by the intruder resulting from inhalation of the particulates, one must first relate the radionuclide concentrations originally in the waste to the radionuclide concentrations in the air breathed by the intruder.

The time delay factor (f_o) is given by the formula $f_o = \exp(-\lambda T)$, where λ is the radionuclide-specific decay constant and T is the period between the time that the disposal facility is closed and the time that the waste is contacted by the intruder. This period is assumed to be zero for this example and so the delay factor is here equal to unity.

The site design factor (f_d) accounts for the potential for uncontaminated soil to mix with the waste during the disposal, so only a fraction of the particulates released into the air is from the disposed waste. The remaining particulates are from the uncontaminated soil. If, in this hypothetical example, it is assumed that the waste is mixed with soil at a ratio of 4 to 1, then f_d is equal to 0.8.

The calculations in this example are based on the assumption that the waste will disperse into the air in a similar manner as the site soil. However, the waste will probably be much less dispersible than ordinary soil. This is accounted for by the waste form and package factor (f_w) which corrects for the relative ability of the waste to disperse into the air as respirable particulates. It is assumed in this hypothetical example that the dispersibility of the waste is ten times less than that of natural soil. Thus, (f_w) for this example is equal to 0.1.

The site selection factor (f_s) accounts for the tendency for construction operations to raise dust at the site as well as the fraction of a year that the worker works at the site. The "tendency for construction operations to raise dust" could depend upon a number of variables such as the type of construction activity performed or the silt content of the soil. Equations have been developed to approximate this, however, and for this example assume that this parameter is determined to be equal to 10^{-6}. The fraction of a year that the intruder works at the site is assumed to be equal to 0.057 (500 hrs/8760 hrs). The site selection factor for this hypothetical case is therefore equal to 5.7×10^{-8}.

Putting the four factors together, one arrives at a value for (I) equal to 4.56×10^{-9}. In other words, the intruder in this case is breathing over the course of his construction activity an average radionuclide concentration in air equal to the radionuclide concentration in the waste (C) times I. Assuming that all waste received at the site contains an average concentration of a given radionuclide equal to 1 Ci/m^3, the radionuclide concentration in air averages 4.56×10^{-9} Ci/m^3 during the time that the intruder is at the site.

There are of course a number of scenarios which may be used calculate release and transport of radionuclides at different points in time as waste is transported, treated, and disposed. However, practically all release/transport scenarios may be structured in a similar manner. Throughout this report, the above four factors are repeatedly referenced although the calculational details of each factor will vary considerably depending upon the particular scenario considered.

2.3.3 Pathway Dose Conversion Factors

Once the concentration of a given radionuclide is determined at a given biota access location, the interaction of the radionuclide (and the radiation it emits) with human tissue must be quantified and the resultant dose rate determined. This is done using a series of radionuclide specific parameters known as pathway dose conversion factors (PDCF) which are independent of the original means of contamination. To illustrate, consider the previous example of the construction worker who is immersed in a cloud of air containing a known concentration of a particular nuclide (Ci per m^3 of air). Potential exposures to the worker could result from the following five pathways:

- inhalation of the contaminated air;
- direct ionizing radiation exposure from being immersed in the contaminated air;
- direct ionizing radiation exposure from contaminated dust which has settled to the ground surface;
- inhalation of contaminated dust which has been resuspended from the ground surface; and
- immersion in the contaminated dust which has been resuspended from the ground surface.

The resulting impacts would be determined by a series of equations describing the movement and uptake of the radioactivity. Factors such as the breathing rate of the individual would be considered, in addition to the resuspension rate of the deposited radionuclides, and so forth. A series of dose conversion factors (termed in this report fundamental dose conversion factors (DCF) to distinguish them from PDCF's) would be used to determine resulting exposures. (For example, once a known quantity of a given radionuclide is inhaled, a dose rate can be determined by multiplying by an appropriate inhalation DCF.) The total dose rate would be determined by adding the dose rate obtained from all five pathways.

Consider, however, that one may be interested in calculating dose rates to several individuals at different times. In addition, the same five pathways would be appropriate for other disposal facilities. It would be highly inconvenient to repeatedly calculate radionuclide movement and uptake through the five example pathways and repeatedly sum the result. What is done, then, is to create a single pathway dose conversion factor for a given radionuclide and organ of interest which is a combination of the five example pathways. That is, given a unit concentration (1 Ci/m^3) of a radionuclide in air, dose rates from all five pathways are summed to form a single effective "pathway dose conversion factor." Different pathway dose conversion factors may be created for different combinations of environmental pathways.

The pathway dose conversion factors used in this report are described in Tables 2-19 and 2-20. All PDCF's, except PDCF-8, are given in units of mrem/yr per Ci/m^3 in the media at the biota access location. Depending upon the situation, the media may consist of a solid (Ci per m^3 of soil), a liquid (Ci per m^3 of water), or gas (Ci per m^3 of air). PDCF-8, which is a new PDCF formulated for this report, is given in units of mrem/yr per Ci/m^2. For PDCF-8, the biota access location is contaminated ground surface. It has been included in order to calculate impacts resulting from a specific scenario involving inadvertant intrusion into a closed disposal facility; namely, a source imbedded in the wall of an excavation.

PDCF's are calculated for all 100 radionuclide/solubility combinations considered in this report, and for 9 different individual human body organs. The nine organs considered include the whole body, lung, kidney, liver, red bone marrow, bone surface, stomach wall, lower large intestine, and thyroid. An effective whole body equivalent based upon ICRP-26 and ICRP-30 methodology is also calculated for each radionuclide and appropriate solubility class. This is the same approach as that used by the authors in reference 9.

Different PDCF's are used appropriate to the given situation. For example, PDCF-6 would be used to calculate impacts to persons using water withdraw from a well, while PDCF-7 would be used to calculate impacts to persons withdrawing water from a surface stream. Some PDCF's are composed of primary and secondary pathways.

The PDCF's used in this report are generic. That is, they are constructed assuming a set of parameters typical of most environmental conditions and human actions. The details of the pathway dose conversion factors, including the calculational formulas, environmental transfer factors, and fundamental dose conversion factors assumed (which are obtained using the ICRP-30 methodology) are given in Appendix D of this report. The PDCF's

TABLE 2-19. Pathway Dose Conversion Factor Components

PDCF	Biota Access Media	Uptake Pathways	
1*	Air	Inhalation (air)	(p)**
		Direct Radiation (air)	(p)
		Inhalation (soil)	(s)**
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
2*	Air	Inhalation (air)	(p)
		Direct Radiation (air)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
3	Air	Inhalation (air)	(p)
		Direct Radiation (air)	(p)
		Food (air)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
4	Soil	Food (soil)	
5	Soil	Direct Radiation (volume)	
6	Well Water	Food (water)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
7	Open Water	Food (water)	(p)
		Fish (water)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
8	Soil	Direct Radiation (area)	(p)

(*) PCDF=1 is used for exposures that last approximately for an entire year for several years (chronic exposures), while PDCF-2 is used for exposures that occur only once for considerably less than a year (acute exposures).

(**) (p) = primary pathway, (s) = secondary pathway.

TABLE 2-20 . Access Location-to-Human Pathway Descriptions

Pathway Designation	Description
Food (soil)	This uptake pathway includes a total of three subpathways and denotes uptake of radionuclides originating in plants via soil-to-root transfer from contaminated soil: plant-to-human plant-to-animal-to-human plant-to-animal-to-product-to-human
Food (air)	This uptake pathway includes a total of six subpathways and includes the above three food (soil) subpatways resulting from uptake of radionuclides originating on plant surface via deposition from contaminated air and the same three food (soil) subpathways resulting from fallout contamination of the ground.
Food (water)	This uptake pathway includes a total of nine subpathways and includes all the food (soil) pathways resulting from radionuclides originating on plant surfaces via irrigation deposition from contaminated water and from irrigation contamination of the ground. The following three subpathways in addition to the plant pathways are added: water-to-human water-to-animal-to-human water-to-animal-to-product-to-human
Ingestion (fish)	Uptake of radionuclides from eating fish caught in contaminated open water.
Inhalation (air)	Uptake of radionuclides from breathing air contaminated due to suspension of contaminated soil particulates by human activities.
Inhalation (soil)	Uptake of radionuclides from breathing air contaminated due to natural suspension and volatilization of surface soil.
Direct Radiation (volume)	Direct exposure to ionizing radiation from standing on ground homogenously contaminated.
Direct Radiation (area)	Direct exposure to ionizing radiation from standing on ground whose surface is contaminated.
Direct Radiation (air)	Direct exposure to ionizing radiation from standing in air homogenously contaminated.

are calculated from environmental transfer and other parameters within the computer codes, which represent a departure from the original Part 61 analysis methodology in which PCDF's were determined elsewhere and merely read as a series of tables. This revised approach allows for greater flexibility in incorporating updated parameter values or parameter values which better reflect site-specific or region-specific environmental conditions. Two region-specific transfer parameters are in fact considered in this report. These include parameters relating to the irrigation rates of crops and to the consumption of plants by animals (See Appendix D).

2.4 General Structure of Computer Codes

This section presents a brief overview of the system of codes developed in this report. Details of the codes, including the listings of the source codes and the data files, are presented in Volume 2 of this report.

Two major and a number of minor codes were developed in this effort. The major codes consist of the following:

CLASIFY: Given the physical/chemical/radiological characteristics of waste streams and the waste processing options, this code classifies the waste streams into the three classes defined in 10 CFR Part 61 (A, B, and C), as well as a new hypothetical class (D) developed for this report, and provides input to the IMPACTS code.

IMPACTS: Based on input from CLASIFY, specific disposal technology information and disposal site environmental characteristics, this code calculates several of the impact measures considered in this report including groundwater migration and overflow impacts, intrusion and exposed waste impacts, and exposures from potential operational accidents.

There are four minor codes considered in this report. These codes are as follows:

INVERSE: Based on a given set of dose limitation criteria, disposal site environmental characteristics, and specific disposal technology information, this code determines the maximum allowable concentration or activity limits for disposal of wastes.

INTRUDE: Based on physical/chemical/radiological characteristics of waste streams, this code calculates the radiological impacts from inadvertant intruder scenarios as a function of time.

VOLUMES: Based on waste stream specific volume generation rates and regional distributions of generator facilities, this code projects waste stream specific region-dependent annual volumes.

ECONOMY: Based on waste stream specific annual volumes disposed and specific disposal technology information, this code calculates disposal costs per unit volume of waste. It also calculates waste processing and transportation costs, radiological impacts, and occupational exposures; disposal facility occupational exposures; and land use.

The major codes are discussed in Sections 2.4.1 and 2.4.2, while the minor codes are considered in Section 2.4.3.

2.4.1 CLASIFY

The main purpose of this code is to provide input to the IMPACTS code. It classifies the waste streams into the four waste classes being considered (A, B, C, and D), evaluates and tests a number of waste processing options (waste spectra), and prepares the waste stream specific information in a format suitable for processing by the IMPACTS code. Since the waste classification process is independent of waste stream volume projections, a given output of CLASIFY can be used in several runs of IMPACTS.

From a system oriented point of view, CLASIFY combines the multitude of alternatives represented by two different data files (WASCAR.DAT and LIMITS.DAT) in a manner specified by a third file (CLACON.DAT) into a single output file (CLAOUT.DAT) for use by IMPACTS.

CLASIFY always uses a unit magnitude for each waste stream considered, and allocates the waste stream into fractions if it has a concentration or an activity distribution. A unit volume (1 m^3) is considered for a routine or activated metal waste stream, while a single source is considered for a source waste stream. A brief discussion of the input data and code mechanics are presented below.

2.4.1.1 Input Data

As stated, there are three files that provide input data to this code: LIMITS.DAT, WASCAR.DAT, and CLACON.DAT. These are considered below.

LIMITS.DAT contains 100 records of information with each record representing a specific radionuclide/solubility combination (see Table 2-7). Each record provides data on the radionuclide name and solubility, plus the radionuclide concentration and activity limitations for four classes of waste (A, B, C, and D) -- that is, four concentration limits and four activity limits are specified. It also specifies area concentration limits in units of Ci/m^2. In addition to these, an index called NCLX is assigned to each radionuclide. Some of the radionuclides are listed in Tables 1 and 2 of 10 CFR Part 61.55 and are specifically used to determine the classification status of wastes. Optional values of the NCLX index are given in Table 2-21.

WASCAR.DAT may contain as many records as the code user specifies. Each record represents a specific waste stream, and provides waste stream specific information including the waste stream name, the radionuclides present in the waste stream and their concentrations (or activities, as applicable), the waste processing options to be considered (including the ISPC indices and densities), the total concentration or activity distribution of the waste stream, if any, whether the waste stream is to be classified using Part 61 procedures or different, usually more stringent, procedures, and other waste stream specific information.

TABLE 2-21 . Definition of NCLX Index Values

NCLX Value	Part 61 Tables	Comments	Example
0	None	Nuclide not in Tables	U-238
1	1	Limit in nCi/g or Ci	Pu-239
2	1	No metal extension	I-129
3	1	Metal extension	C-14
4	2	No metal extension	Cs-137
5	2	Metal extension	Ni-59
6	2	Half-life < 5 years	Fe-55

CLACON.DAT file stands for CLAsify CONtrol file and is the most important, albeit the smallest, of the files used by CLASIFY. This file specifies the manner in which the waste streams given in WASCAR.DAT are to be considered, and whether the disposal technologies being considered in IMPACTS can stabilize the waste streams. (That is, the code user may optionally use disposal technology design rather than waste processing to achieve waste stability.) It also may contain overriding information for the waste stream specific characteristics read from WASCAR.DAT (e.g., whether to encapsulate the waste stream in a high integrity container).

2.4.1.2 Function and Output

As stated, the code combines the multitude of alternatives presented by four data files into one file for use by the IMPACTS code. This is the major reason for CLASIFY -- that is, consolidating an almost unmanageable number of alternatives into a problem which can be managed. Another reason for CLASIFY is that it permits experimentation with alternative limits for source waste streams as well as with Class D streams.

Its output, which is called CLAOUT.DAT, is structured in a manner similar to WASCAR.DAT -- that is, each record contains information on a single waste stream. However, in addition to the information retrieved from WASCAR.DAT and passed through (such as the names and waste stream densities), it also contains information on the waste class, the fraction of the waste stream volume in that class and modified radionuclide concentrations or activities (if the waste stream has a concentration or activity distribution), waste form behavior indices, and waste processing indices and costs. The data retrived from distributions, if any, is incorporated into the specific radionuclide concentrations or activities.

2.4.2 IMPACTS

The function of IMPACTS is to determine several of the impact measures considered in this report for a given combination of (1) waste streams and processing options, (2) disposal technology alternatives, and (3) disposal site environmental settings. The impact measures considered in this code include groundwater migration and overflow impacts, intrusion and exposed waste impacts, and exposures from potential operational accidents. Disposal costs and other impact measures are considered in the code ECONOMY (see below).

2.4.2.1 Input Data

In addition to the CLAOUT.DAT file provided by the code CLASIFY, IMPACTS uses five other data files: IMPCON.DAT, DISTEC.DAT, ENVIRO.DAT, FUNDCF.DAT, and VOLUME.DAT. These files are considered below.

IMPCON.DAT, which stands for IMPacts CONtrol file, specifies the problem to be considered including the general facility and cost indices (see Section 2.2.1), the disposal configuration (see Section 2.2.3), and the waste stream specific information discussed in Section 2.2.2. It also may contain overriding information on some of the disposal technology characteristics (retrieved from DISTEC.DAT) and environmental characteristics (retrived from ENVIRO.DAT).

DISTEC.DAT may contain as many records as the code user specifies. Each record represents a specific disposal technology, and is accessed using the ID index (see Section 2.2.3). Each record contains disposal technology parameters EFF, SEF, DPT, and DTK, as well as parameters required to calculate disposal costs.

ENVIRO.DAT may also contain as many records as the code user specifies. Each record represents a different environmental setting, and is accessed using the IR index (see Section 2.2.1). Each record includes the parameter values required in impact calculations (see Volume 2). In this report, the number of records have been kept to 4 reference settings (see Section 2.1). Region dependent uptake factors (see Appendix D) are also included in this file.

FUNDCF.DAT contains 100 records with each record representing a specific radionuclide (see Table 2-7). Each record contains information on the fundamental dose conversion factors, retardation coefficients, radionuclide partition ratios, etc. The information retrieved from this file can not be altered by any other input. If the code user wishes to use a different set of values, the file itself must be altered.

Finally, VOLUME.DAT may also contain as many records as the code user specifies. Each record represents a specific waste stream, and is accessed using the waste stream name (see Section 2.2.2.3). Each record gives the projected volumes (or number of sources) to be generated between the years 1981 and 2030, inclusive, for the four waste generating regions considered in this report. This file is generated by the code VOLUMES discussed in Section 2.4.3.

2.4.2.2 Function and Output

The function of this code is to produce most of the radiological impact measures considered in this report given the combination of (1) waste streams and processing options, (2) disposal technology alternatives, and (3) disposal site environmental setting. Its output is primarily as hard-copy, i.e., printed computer output.

2.4.3 Other Codes

These codes consist of INVERSE, INTRUDE, VOLUMES, and ECONOMY. The first two perform some specific functions. The latter two were written primarily to reduce the complexity of the IMPACTS code, and to help isolate the uncertainties in the parameter values assumed in this report.

2.4.3.1 INVERSE

This code calculates the maximum radionuclide concentrations or activities (as applicable) allowable in a specific waste stream in combination with a specific disposal technology located at a specific environmental setting, i.e., the inverse calculation discussed in Section 2.3.1. It requires organ-specific individual dose limitation criteria to be specified in order to produce the limiting values. Its output may be used to specify the radionuclide concentration and activity limits contained in the file LIMITS.DAT (see above).

This code uses four data files: INVCON.DAT, FUNDCF.DAT, DISTEC.DAT, and ENVIRO.DAT. All these files, except INVCON.DAT, have been explained above. Further details on their use by INVERSE can be found in Volume 2 of this report.

INVCON.DAT stands for INVerse CONtrol file, and contains the dose limitation criteria, the disposal technology to be considered, the environmental setting in which the disposal site is located, and the waste form behavior indices for the hypothetical waste stream. The code then calculates the concentrations (or activities for source waste streams) that would result in the specified dose limitation criteria for all the 100 radionuclides considered from a number of impact scenarios.

2.4.3.2 INTRUDE

This code is very similar to the code IMPACTS in that it uses most of the same data files, and calculates radiation exposures. However, INTRUDE calculates only radiological impacts associated with the intruder initiated scenarios (see Section 4.2) as a function of time. It does not calculate the other impact measures. This code is designed to enable assessment of the short- and long-term implications of different combinations of waste streams disposed at the same location. Further details of this code can be found in Volume 2 of this report.

2.4.3.3 VOLUMES

The rationale for this code was examined in Section 2.2.2.3. It isolates the uncertainties which may be associated with waste stream projections. This code can be repeatedly run to update these projections without affecting the remaining codes. It uses two input files, called VRATES.DAT and REACTR.DAT, and produces the VOLUME.DAT file. VRATES.DAT file contains basic generation rates for the waste streams, their rate of increase, etc. REACTR.DAT file contains data on all the reactors either operating or projected to operate within the time period of interest (1981 to 2030): their names and locations, electrical and thermal power generation rate, and start-up and projected decommissioning dates. Further details of this code can be found in Volume 2 of this report.

2.4.3.4 ECONOMY

This code is primarily designed to reduce the complexity of the IMPACTS code, and to isolate some of the uncertainties associated with the disposal technology construction procedures and costs. It is identical with the code IMPACTS at the front end, e.g., it uses all of the same input files: CLAOUT.DAT, IMPCON.DAT, DISTEC.DAT, ENVIRO.DAT, FUNDCF.DAT, and VOLUME.DAT; however, a considerable amount of the information contained in the above files is ignored. It also uses a file called WASPAC.DAT (see below). It calculates disposal technology costs per unit volume of waste and other impact measures using the information contained in Appendix C, i.e., the DISTEEC.DAT file. It also calculates waste processing and transportation costs, radiological impacts, and occupational exposures; disposal facility occupational exposures; and land use. Further detais of this code can be found in Volume 2 of this report.

WASPAC.DAT may contain as many records as the code user specifies. Each record represents a specific waste package distribution among a set of reference waste packages, and is accessed using the IPK index, the first digit of the waste processing parameters index, I1. However, in practice, it contains the information given in Table 3-9.

CHAPTER 2.0 REFERENCES

(1) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Draft Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0782, September 1981.

(2) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Final Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0945, November 1982.

(3) U.S. Nuclear Regulatory Commission, "10 CFR Parts 2, 19, 20, 21, 30, 40, 51, 61, 70, 73, and 170: Licensing Requirements for Land Disposal of Radioactive Waste, Final Regulation," Federal Register, 47 FR 57446, December 27, 1982.

(4) Wild, R.E., et al., "Data Base for Radioactive Waste Management, Volume 2, Waste Source Options Report," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-1759, November 1981.

(5) Oztunali, O.I., et al., "Data Base for Radioactive Waste Management, Volume 3, Impacts Analyses Methodology Report," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-1759, November 1981.

(6) U.S. Atomic Energy Commission, Directorate of Regulatory Standards, "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," USAEC Report WASH-1238, December 1972.

(7) U.S. Nuclear Regulatory Commission, Division of Waste Management, "Low-Level Waste Branch Technical Position on Waste Form, May 1984.

(8) Oztunali, O.I. et al., "Influence of Leach Rates and Other Parameters on Groundwater Migration," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-3130, September 1982.

(9) Oztunali, O.I. and G.W. Roles, "De Minimis Waste Impacts Analysis Methodology," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-3585, February 1984.

3.0 WASTE PROCESSING AND TRANSPORTATION IMPACTS

This chapter discusses the calculational procedures utilized to determine the impact measures associated with waste processing and transport to a disposal facility. The impact measures considered in this chapter consist of (1) the incremental costs associated with processing and transportation, (2) the population doses resulting from incineration and transportation, and (3) the occupational exposures associated with processing and transportation (including waste loaders and truck drivers). Impact measures associated with unloading and emplacement of the waste at the disposal facility are considered in Chapter 4.

Much of the methodology presented in this chapter is similar to the original Part 61 analysis methodology presented in references 1 through 4. However, costs have been updated to account for inflation, and the input data has been reorganized to expedite incorporation of additional information. Sections 3.1 and 3.2 present the assumptions and methodology associated with the calculation of impact measures associated with waste processing and transportation, respectively.

3.1 Waste Processing Impacts

There are a large number of variations in the parameters associated with waste processing that affect the calculation of impacts. For example, the variations in the processing technologies applied to a given waste stream include the volume reduction process type, the volume increase process type, the location of the processing, the environment in which the processing takes place, and unit impact measures such as costs and person-hours per unit volume of waste processed. Moreover, for each alternative waste processing option, the waste form behavior indices of the final product may be different (see Section 2.2.2).

This section considers impacts associated with waste processing. The waste processing index, I1, is used extensively in these calculations. Alternative parameter values for this index, which is composed of six subindices, are summarized in Table 3-1.

There are two conceptually distinct types of impacts associated with waste processing: population exposures and other impacts. These are discussed in the following sections.

3.1.1 Population Exposures

For the purposes of calculation of population exposures arising from waste processing, only incineration is assumed to result in significant atmospheric releases to the environment, i.e., IPR index values of 5 through 7 (see Table 3-1). The fraction of the radioactivity released depends on the type of incincerator used, the controls on the off-gas system, and the radionuclide.

TABLE 3-1 . Waste Processing Parameters Indices

Symbol	Property	Optional Values
IPK	Packaging Index	Record number of file containing distribution
IPR	Processing Index	0 = No volume reduction 1 = Regular compaction 2 = Improved compaction 3 = Hydraulic press 4 = Evaporation 5 = Pathological incineration 6 = Small calciner 7 = Large calciner
ISL	Solidification Index	0 = Unsolidified waste form 1 = Solidification scenario A 2 = Solidification scenario B 3 = Solidification scenario C 4 = High integrity container (HIC) packaging 5 = Stabilization through another means
ILC	Processing Location	0 = No processing 1 = Processing at the generator 2 = Processing at the disposal facility
IEN	Processing Environment	0 = No incineration 1 = Urban environment 2 = Rural environment
IRE	Processing Radiation Environment	1 = High facility background radiation environment 2 = Low facility background environment radiation

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There are three major assumptions associated with population exposure calculations: (1) the fraction of the total activity released to the environment, (2) the environment that is affected by the processing, and (3) the pathway dose conversion factors used. The following equation is used to calculate the population exposures from incineration:

$$P = \sum_n Q_n f_{rn} \text{ POP PDCF-1} \qquad (3\text{-}1)$$

where

P = Population exposure in person-mrem/year,
n = Radionuclide,
Q_n = Total activity of radionuclide n within incinerated waste (Ci/year),
f_{rn} = Radionuclide dependent release fractions (dimensionless),
POP = Site specific population weighted atmospheric dispersion factor in units of person-year/m^3, and
PDCF-1 = Pathway dose conversion factor in mrem/year per Ci/m^3.

This equation, which is obtained from reference 5, is self-evident except for the factors f_{rn} and POP. PDCF-1 is the chronic exposure factor for airborne biota access locations as discussed in Appendix D. The factors f_{rn} and POP are explained below.

Release Fractions - f_{rn}

The assumed fractions of the total input activity of nuclide n released to the atmosphere, f_{rn}, are given in Table 3-2.

TABLE 3-2. Release Fractions by Incinerator Type

Nuclide	Pathological	Calciner
H-3	0.90	0.90
C-14	0.75	0.25
Tc-99	0.01	0.001
All I and Ru	0.01	0.001
All Others	0.0025	0.000025

A calciner/incinerator (IPR=6 or 7) is generally assumed to have better off-gas controls than a pathological incinerator (IPR=5). Most of the incinerated tritium is released as water vapor. Although some of the tritiated water vapor may condense and deposit in very close vicinity of the release point, this effect is conservatively not considered in this report. Carbon-14 is usually released as tagged combustion gases such as CO and CO_2, and Tc-99, Ru-103, Ru-106, and I-129 are usually considered as

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semi-volatile radionuclides that are harder to control than particulates. All other radionuclides are assumed to be particulates, and particulate release fractions are applied.

These fractions (i.e., f_{rn}) are also used in modifying the final waste product (i.e., disposed waste) concentrations for tritium and carbon-14. Release fractions for other radionuclides are conservatively assumed not to affect the concentrations of the final waste product.

Site Selection Factor - POP

The factor POP corresponds to the site specific population weighted atmospheric dispersion factor (i.e., X/Q weighted by the population distribution). The user of the codes may choose to provide site specific values for the calculation of POP, e.g., using the fraction of a year that the site is in a given stability class, the average wind speed within the stability class, and the population distribution for the specific site (see Ref. 5 and Volume 2 of this report). In the absence of site specific input, the following generic values and methodology are utilized.

The population distribution may be one of the generic rural environment distributions as presented in Table 3-3 for each reference site (Ref.1 through 5). For urban environments (IEN=2), the population distributions are multiplied by a factor of 10.

TABLE 3-3 . Population Distributions for Reference Site Environments

Distance From Source	Northeast	Southeast	Midwest	Southwest
0-5 miles	3,440	2,024	3,070	59
5-10 miles	20,513	8,115	4,998	180
10-20 miles	73,636	36,000	27,890	3,529
20-30 miles	121,559	124,995	104,181	9,062
30-40 miles	556,639	203,435	121,893	4,888
40-50 miles	1,012,788	104,933	359,146	27,158

The second component in the calculation of POP is the atmospheric dispersion factor (X/Q) which is calculated using the following equation:

$$(X/Q) = 2.032 \exp[-(1/2)(h/\sigma_z)^2]/[u\ r\ \sigma_z] \qquad (3\text{-}2)$$

where

h = effective height (m) of release above ground surface,
σ_z = standard deviation of the plume concentration distribution in the vertical direction (m),

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u = mean wind speed in the sector at the point of release (m/sec),
r = distance from the source point (m).

The vertical standard deviation of the plume (σ_z) is given as a function of distance and stability class in many references. One form for this factor is (Ref. 6):

$$\sigma_z = (ar)\,(1 + br)^c \tag{3-3}$$

where r is the distance from the release point, and a, b, and c, are constants that depend on the stability class. Six stability classes in order of increasing stability are characterized based on criterion stated by Pasquill. Values for the above constants for each of the stability classes are given in Table 3-4 (Ref. 6).

TABLE 3-4 . Vertical Standard Deviation Constants

Pasquill Stability Class	Constants a	b	c
A	0.2	0	0
B	0.12	0	0
C	0.08	0.0002	-0.5
D	0.06	0.0015	-0.5
E	0.03	0.0003	-1
F	0.016	0.0003	-1

Two general situations may be envisioned which involve airborne release: waste incineration and dispersion of contaminated soil. The latter situation always involves ground level release (i.e., h = 0 m), while the former situation requires assumptions to be made regarding the height of the incinerator stack. Given the uncertainties regarding the design of incinerators, ground level release is conservatively assumed in this case. In any case, the effects of release height are generally not significant further than about 1000 m from the release point. The topography of the area is more significant.

To calculate POP, a (X/Q) value is calculated for the midpoint of each population distribution ring given in Table 3-3. This value is then multiplied by the population within each ring, and the resulting products summed. Specifically, generic stability classes are assumed in this report for site conditions consisting of 1/3 Stability Class C with wind speed of 3 m/sec, 1/3 Stability Class D with wind speed of 3 m/sec, and 1/3 Stability Class F with wind speed of 2 m/sec. Under these conditions, assuming ground level release and incorporating equation 3-3 into 3-2, equation 3-2 reduces to the following:

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$$(X/Q) = 4.168 \times 10^{-8}\, q(r)/r^2 \qquad (3\text{-}4)$$

where (X/Q) is in units of yr/m^3, r is the distance from the source point in m, and q(r) is given by the following:

$$q(r) = .133(1+.0002r)^{1/2} + .178(1+.0015r)^{1/2} + 1 + .0003r \qquad (3\text{-}5)$$

Using these assumptions, assuming rural environments, and using the reference distances and population distributions given in Table 3-3, the POP values given in Table 3-5 are obtained for each of the four reference sites. POP values assuming urban environments (IEN=2) are all a factor of 10 higher.

TABLE 3-5 . Reference Population Factors ($person\text{-}year/m^3$)

Site Environment	POP
Northeast	5.05×10^{-10}
Southeast	1.75×10^{-10}
Midwest	1.93×10^{-11}
Southwest	1.33×10^{-11}

3.1.2 Other Impacts

Other waste processing impacts are calculated based on the unit cost rates and labor hours that have been assumed based upon information presented in references 3, 4 and 7 for selected waste processing options. These unit rates are summarized in Table 3-6.

The 1980 cost rates obtained from the original references have been updated to the 1984 cost rates based on an assumed inflation factor of 1.23. This inflation factor has been obtained from comparing the producer price index for finished durable goods (excluding food) for the respective years (Ref. 8).

Occupational exposures resulting from waste processing occur primarily as a result of repair and maintenance of the waste processing equipment. Requirements for repair and maintenance are highly variable due to the wide variation in the design of processing equipment, as well as variations in the effectiveness of administrative controls at waste generator facilities. This presents difficulties in estimating generic exposures resulting from equipment repair and maintenance.

TABLE 3-6 . Summary of Processing Unit Impact Rates

Process	Costs (1980 $)	Costs (1984 $)	Labor (hours)	Units
Compaction				
Regular	335	412	15	Per m^3
Improved	503	619	15	of
Hydraulic Press	1006	1237	15	Input
Evaporation	690	849	4.42	Per m^3 of Input
Incineration				
Pathological	2060	2534	8	Per m^3
Small Calciner	1938	2384	6.12	of
Large Calciner	1039	1278	5.35	Input
Solidification*				
Scenario A	1282	1577	4	Per m^3
Scenario B	2445	3007	4	of
Scenario C	3056	3759	4	Output

* Costs for producing a structurally stable waste form are assumed to be 20% higher.

In this report, the occupational exposures are assumed to be independent of the waste concentrations, and are calculated as the product of the labor hours required to process a unit volume of waste and the radiation field associated with the general work environment. The person hours required to process a unit volume of waste is substantially more than the repair and maintenance hours; however, the repair and maintenance hours may be assumed to be proportional to the volume of waste processed, i.e., increased volume processed results in increased repair and maintenance requirements. The radiation fields associated with the general work environment, on the other hand, are likely to be less than the radiation fields associated with repair work. The radiation fields assumed in this report may be taken to represent an average of those for repairing and maintaining the equipment and those for routine processing.

In this report, a subindex (IRE) of the waste processing index (I1) has been designated to denote the waste processing radiation environment. This index is used to retrieve a value from an array called RADF in the code (read from the file ENVIRO.DAT), which is multiplied by the above discussed labor hours to obtain the occupational exposures. The generic radiation fields assigned to alternative processing radiation environments are presented in Table 3-7.

TABLE 3-7 . Assumed Generic Radiation Fields

IRE	Radiation Field
1	0.5 mR/hr
2	0.1 "

In this report, an IRE value of 1 is used for waste streams processed in a nuclear power plant, while an IRE value of 2 is used for waste streams processed in most non-fuel cycle facilities. As stated, the codes read these values from a file called ENVIRO.DAT, and the code user has the option of altering these radiation environments or using the IRE values 3 through 9.

3.2 Transportation Impacts

This section discusses the calculational procedures used to determine impacts associated with transportation of waste to the disposal facility. Section 3.2.1 presents the packaging and shipping assumptions utilized in the calculations. Transportation costs, population exposures, and occupational exposures are presented in Sections 3.2.2 through 3.2.4, respectively.

It should be stressed at the outset of this section that the estimated costs, population exposures, and occupational exposures are valid only for comparison purposes. That is, they can be used to compare the relative benefits and costs of various alternatives; they do not necessarily correspond to actual costs and exposures.

3.2.1 Packaging and Shipping Assumptions

In order to estimate impacts associated with waste transportation (e.g., occupational exposures, population exposures, and costs), data are necessary on three major aspects: (1) waste packaging parameters, i.e., container types and sizes, (2) the radiation levels at the surface of the waste packages, and (3) the shipment mode (vehicles and overpacks used). These three aspects are interrelated to a certain degree. For example, packaging parameters strongly influence the radiation levels and vice versa, and the radiation level determines the shipment mode. In addition, all these three aspects vary over a wide range of values. Consequently, some simplifying assumptions regarding waste packaging and transportation are made in this report based upon past experience.

Waste Packaging Parameters

There are many different types of packaging currently utilized for shipment and disposal of waste. These packages include wooden boxes of

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various sizes ranging from 10 ft^3 to 248 ft^3, 55 gallon drums, and liners (usually carbon steel) of various sizes ranging from 16 ft^3 to 200 ft^3 which fit into transport casks. Given the generic type of analyses performed in this report, these packages are generalized into the five different categories given in Table 3-8.

TABLE 3-8 . Generic Packaging Parameters

Package	Volume (ft^3)	Costs ($ 1984) Regular	Costs ($ 1984) HICs
Large wooden boxes	128	500	6000
Small wooden boxes	16	250	4000
55-gallon drums	7.5	25	350
Small liners	50	4000	4000
Large liners	170	5000	5000

The primary rationale for selecting these package sizes is that they appear to be the most widely used sizes, and may be used to represent an average of other packages. For example, the 128 ft^3 box is the most commonly used (4'x4'x8') size to ship low specific activity (LSA) waste, the 170 ft^3 liner is the commonly available 6'x6' right-circular cylindrical resin tank, etc. A number of waste package distributions have been assumed using available shipping and survey data, and are presented in Table 3-9.

TABLE 3-9 . Generic Package Distributions (percents)

Packaging Index - IPK	Large Boxes	Small Boxes	Drums	Small Liners	Large Liners
1	0	0	69	15	16
2	23	8	69	0	0
3	0	3	97	0	0
4	50	0	50	0	0
5	100	0	0	0	0
6	0	100	0	0	0
7	0	0	100	0	0
8	0	0	0	100	0
9	0	0	0	0	100
10	0	0	50	0	50
11	0	90	10	0	0
12	0	0	15	15	70
13	28	16	56	0	0
14	10	5	85	0	0
15	70	30	0	0	0

Each of these distributions are stored in a single record of a file called WASPAK.DAT, and the record number of the file corresponds to the packaging index (IPK - see Table 3-1) which is stored in the ISPC array. During the transportation analysis, the proper distribution is retrieved for each waste stream. The WASPAK.DAT file can easily be expanded if the code user wishes to use another distribution, and the record number of the file assigned to the appropriate waste stream.

For sealed sources (i.e., I10 > 0), it is assumed that there is one source per package. Moreover, in the analyses performed in this report, it is assumed that this package is a 55-gallon drum (i.e., IPK=7). The user can alter this assumption through the modification of the ISPC indices.

Surface Radiation Levels

Radiation levels at the waste package surfaces affect the care required in handling of wastes and the shielding that may be required during transportation. Depending on the package size involved and the total activity content of each package, different waste packages have different surface radiation readings. Moreover, package sizes and packaging procedures are instrumental in determining the self-shielding afforded by some of the waste packages. However, there can be significant variations in the level of care required for each waste package due to variations in specific activities within a given waste stream.

For the purposes of this report, the waste streams are generically classified into the same three categories used in the original Part 61 analysis methodology according to the level of care required to handle each waste stream. These care levels and the associated radiation levels are presented in Table 3-10.

TABLE 3-10 . Care Level Categories

Designation	Radiation Level
Regular care	<10 mR/hour
Special care	10-1000 mR/hour
Extreme care	>1 R/hour

Part of the rationale for these three categories is based on the U.S. Department of Transportation limit of 10 mR/hr on the surface of a shipping vehicle. Regular care wastes can be shipped in a one-way transportation mode, i.e., no return trip for the shielding cask or trailer is required. The other part of the rationale for these categories is based on the handling requirements for wastes at the disposal sites. Waste with radiation levels more than 1 R/hr usually require remote handling techniques.

In the original Part 61 analysis methodology, the care level was assumed to be independent of waste package shape and volume. The care level was

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assumed to depend only on the total specific activity contained in the waste package and the presence or absence of radionuclides emitting high-energy gamma rays.

In this report, however, the care level of the waste is determined using the package radiation level as approximated using the actual radioactive concentrations in each waste stream. The following equation is used to calculate the surface radiation level of the packages:

$$R = (1/8760) \sum_{n} Cn \;\; PDCF\text{-}5 \qquad (3\text{-}6)$$

where

R = Radiation level in mR/hour,
8760 = Conversion factor in hours/year,
n = Radionuclide,
Cn = Concentration of nuclide n in Ci/m^3, and
PDCF-5 = Pathway dose conversion factor in mR/year per Ci/m^3.

This equation is self-explanatory, and conservatively ignores the geometry of the waste containers. It represents exposures to someone hypothetically standing next to a very extensive (width and height) stack of closely arrayed waste containers. PDCF-5 derived from the fundamental dose conversion factor DCF-3 discussed in Appendix D.

As stated, for sealed sources, it is assumed that the shipping package is a drum, and each package contains one sealed source. Consequently, C_n, which is the average total activity per sealed source is distributed over the volume of a drum and the R value calculated. This is an arbitrary approximation. Sealed sources would normally be placed somewhere in the middle of the drum to afford additional shielding. This shielding is, however, somewhat accounted for through distributing the total activity over the volume of the drum.

Shipment Mode

Similar to the numerous different types of available waste packages, there may exist many different shipment modes ranging from rail and barge transport to truck transport. Many different types of overpacks may be used depending on the handling and shielding requirements for individual waste packages.

In this report, only truck transport is considered because trucks are by far the most commonly used mode of transportation and truck transport is radiologically the most conservative case. Vehicles and overpacks utilized in truck shipments depend on package sizes as well as package shielding requirements. In this report, six different types of transport vehicles and overpacks, which were used in the original Part 61 analysis methodology, are assumed. Generic waste shipping modes are presented in Table 3-11.

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TABLE 3-11 . Generic Waste Shipping Modes

Description	Symbol
1-drum shielded casks	1D
Small shielded casks	SC
Large shielded casks	LC
Shielded trailers	ST
Flatbed trailers	FT
Vans	VN

The use of particular types of vehicles and overpacks is strongly influenced by the level of care required for safe waste handling and transport. Vans are assumed to be suitable for all types of containers in the regular care category, with the exception of large liners which require casks. In addition, flatbed trailers are assumed to be used only for large boxes of regular care wastes and packages which do not require additional shielding. Shielded trailers are assumed to be required for large and small boxes and drums of special care wastes. Some of these small boxes and drums, as well as large and small liners, are assumed to require casks. Casks are assumed to be the only acceptable mode of transport for extreme-care wastes.

Large casks are used for transporting either large liners or fourteen 55-gallon drums, while small casks are used for transporting either small liners or six 55-gallon drums. These casks are transported to the disposal facility via flatbed trailers.

The percentage use of different vehicles and overpacks for each container have been estimated considering records of waste shipments and other references (see Appendix B). A tabular listing of the basic assumptions made for the transportation of wastes is presented in Table 3-12. Extreme-care liner shipments and concrete vaults have been assumed to be "overweight" shipments since these require significant shielding for transportation purposes. These are also designated in Table 3-12.

3.2.2 Exposures During Transportation

The occupational and population exposures incurred during transportation are calculated based on total loaded miles, the number of rest stops along the route, and the number of loaded shipments. The concept of loaded miles and shipments allows to be eliminated from consideration those miles in which the vehicle is empty because it is on a return trip.

In a manner similar to the original Part 61 analysis methodology, four waste generating regions are considered, with one reference disposal site within each region. The updated analysis methodology, however, now allows consideration of impacts from waste transported across regional boundaries. Therefore, population exposures incurred in each region are

TABLE 3-12 . Packaging and Shipment Mode Parameters

Care Level and Container	Overpack (a)	Pieces	Percent Volume	Loading Person-minutes Per Container
Regular Care				
Large Box	Van	3	24	120
	FB	4	76	60
Small Box	Van	36	100	12
Drum	Van	70	100	12
Small Liner	Van	11	100	83
Large Liner	LC	1	100	720
Special Care				
Large Box	ST	3	100	180
Small Box	ST	36	96	20
	LC	6	4	150
Drum	ST	70	48	12
	LC	14	51	88
	SC	6	1	156
Small Liner	SC	2	100	360
Large Liner	LC	1	100	720
Extreme Care				
Drum	SC	6	51	156
	1D	1	49	360
Small Liner	SC (b)	2	100	360
Large Liner	LC (b)	1	100	900

(a) FB = Flatbed Trailer; ST = Shielded Trailer; LC = Large Cask; SC = Small Cask; 1D = 1-drum Cask.
(b) These shipments are assumed to be overweight.

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calculated separately. This means that assumptions must be made on the route length within each region as well as rest stops within each region. Table 3-13 presents the inter and intraregional transportation distances as well as the number of rest stops within each region.

TABLE 3-13 . Interregional and Intraregional Shipment Distances (in Miles) and Rest Stops (in Parantheses) for Transportation Exposure Calculations

Generating Region	Disposal Region	Affected Region 1	2	3	4
1	1	300(1)	-	-	-
	2	500(2)	500(2)	-	-
	3	500(2)	-	500(2)	-
	4	500(2)	-	1000(2)	1500(3)
2	1	500(2)	500(2)	-	-
	2	-	400(1)	-	-
	3	-	500(2)	500(2)	-
	4	-	500(2)	1000(2)	1500(3)
3	1	500(2)	-	500(2)	-
	2	-	500(2)	500(2)	-
	3	-	-	1000(2)	-
	4	-	-	1000(2)	1500(3)
4	1	500(2)	-	1000(2)	1500(3)
	2	-	500(2)	1000(2)	1500(3)
	3	-	-	1000(2)	1500(3)
	4	-	-	-	1500(3)

Population and occupational exposures are calculated separately for those resulting during transit, and those resulting from stopovers during the trip. Sections below discuss these exposures.

Population Exposures During Transit

The following equation, based on an approach used in reference 5, is used to calculate the population exposures during transit:

$$P = T \times K \times D_i \times (L_i/V_i) \times TDOZ_i \qquad (3\text{-}7)$$

where

P = Total population exposure in person-mrem,
T = Total number of shipments (no/year),

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K = Source strength in mR-ft^2/hour (see below),
D_i = Population density in region i (people/mi^2),
L_i = Distance in region i (miles/trip),
V_i = Speed of the vehicle in miles/hour, and
$TDOZ_i$ = Dose factor in (miles/ft)^2 (see below).

The total number of shipments, T, is calculated as follows: (1) total volume of waste shipped is allocated into packaging categories in accordance with the index I1 (i.e., Table 3-9), (2) package care level categories are determined based upon waste stream radionuclide contents and Table 3-10, (3) the corresponding package shipping mode is retrieved from Table 3-12, and (4) the number of shipments are calculated based on container volumes from Table 3-8, and containers per shipment from Table 3-12. All fractional loads are conservatively assumed to be full loads in this calculation.

For example, assume that there is 10,000 m^3 of a given waste stream that has been determined to be regular care level. The packaging index (I1) of this waste stream indicates that 50% (176,574 ft^3) is shipped in large boxes (128 ft^3), 20% (70,630 ft^3) in small boxes (16 ft^3), and 30% (105,944 ft^3) in drums (7.5 ft^3), i.e., the IPK record for this waste stream in WASPAK.DAT is 0, 50, 20, 30, 0, 0, 0. The following table gives the results of the calculation:

TABLE 3-14 . Example Calculation of Shipment Numbers

Container	Overpack	Number of Shipments
LB	Van (24%)	42378/(3x128) = 111
	FB (76%)	134196/(4x128) = 263
SB	Van (100%)	70630/(36x16) = 123
DD	Van (100%)	105944/(70x7.5)= 202

Thus, the 10,000 m^3 of regular care level waste in this example requires a total of 699 shipments.

Distances in each region have been given in Table 3-13. Other region dependent parameters, D_i, V_i, and $TDOZ_i$ are given in Table 3-15, and are discussed below.

The speed of the vehicle is assumed to be 50 miles/hour in all four regions. However, if the user wishes to alter this assumption, it can be done so through changing values in the file ENVIRO.DAT.

The term $TDOZ_i$ is an integral discussed in Reference 5 that depends on the assumed exclusion distance from vehicle route during transit (x_{min}). The values assumed above correspond to an exclusion distance of 30 ft for the

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TABLE 3-15 . Transportation Parameters

Region	Population Density	Vehicle Speed	TDOZ
Northeast	2280	50	7.06×10^{-5}
Southeast	610	50	7.06×10^{-5}
Midwest	790	50	7.06×10^{-5}
Southwest	60	50	3.92×10^{-5}

Northeast, the Southeast, and the Midwest regions, and a distance of 100 ft for the Southwest region. The integral for $TDOZ_i$ is presented below:

$$TDOZ_i = 4/(5280)^2 \int_{x_{min}}^{\infty} dx \int_{-\infty}^{\infty} dr \, \exp(-u_a r)B(u_a r)/[r(r^2-x^2)^{1/2}] \qquad (3\text{-}8)$$

where

$TDOZ_i$ = Dose factor in $(\text{miles/ft})^2$,
5280 = Conversion factor (ft/mile),
x_{min} = Lower boundary of exposure distance (ft),
r = Integration variable in ft (see Figure 3.2 of Ref. 5),
u_a = Linear attenuation coefficient for air (0.0097 m^{-1}), and
$B(x)$ = Berger buildup factor given by $1 + 0.95x + 0.35x^2$

The remaining term in equation (3-7) is the source strength term, K, which is a function of the characteristics of the waste being transported. It is the strength of the assumed point source at the center of the vehicle as extrapolated from the radiation level at the surface of the vehicle. It is the smaller of two values, one of which is 1000 mR-ft^2/hour (based on the DOT limit for radiation level at the surface of the vehicle (10 mR/hr) multiplied by 100 used to convert the dose rate to the assumed point source at the center of the vehicle) and the other is given by the following equation:

$$K = 100 \times (p_d/p_w) \times (CF/8760) \times \sum_n C_n \times PDCF\text{-}5 \qquad (3\text{-}9)$$

where 100 is the value (square of 10 ft) used to convert the dose rate to the assumed point source at the center of the vehicle, p_d is the density of soil (1.6 g/cm^3), p_w is the density of the waste, CF is a correction factor to account for the finite extent of the vehicle (see below), 8760 is the value to convert units of PDCF from per year to per hour, C_n is the concentration of radionuclide n in the waste, and PDCF-5 is the dose rate in mrem/year for a slab source of infinite thickness and lateral extent containing 1 Ci/m^3 of the radionuclide n. Selecting the smaller of these two values implicitly assumes that shielding will be employed as necessary to bring the vehicle in compliance with the DOT limit of 10 mR/hour at the surface of the vehicle.

The correction factor, CF, arises from the finite extent of the shipping vehicle. Details of its application are presented in Reference 3. In this report it is assumed to be 0.195 for casks, 0.341 for a dumpster, and and 0.546 for vans and trailers.

Population Exposures During Stops

Population exposures during rest stops are calculated in a manner similar to those during transit. However, some modification of the methodology given in Reference 5 is necessary. The following equation is used.

$$P = T \times K \times D_i \times S_i \times SD_i \times TDZ_i \qquad (3\text{-}10)$$

where

P = Total population exposure in person-mrem,
T = Total number of shipments (no/year),
K = Source strength in mR-ft^2/hour (see above),
D_i = Population density in region i (people/mi^2),
S_i = Number of stops in region i,
SD_i = Duration of stops in region i (hours), and
TDZ_i = Dose factor in $(miles/ft)^2$ (see below).

The parameters P, T, K, and D_i are as explained above. S_i is given in Table 3-13, and SD_i has been assumed to be 1 hour for all regions. The parameter TDZ_i is a modified version of $TDOZ_i$ given in Reference 5. It is given by the following integral:

$$TDZ_i = 2\pi/(5280)^2 \int_{r_{min}}^{\infty} \exp(-u_a r)B(u_a r)dr/r \qquad (3\text{-}11)$$

where

5280 = Conversion factor (ft/mile),
r_{min} = Minimum exclusion radius (meters),
u_a = Linear attenuation coefficient for air (0.0097 m^{-1}), and
$B(x)$ = Berger buildup factor given by
$1 + 0.95\,x + 0.35\,x^2$

This integral was analytically evaluated for various r_{min} values. The results are given in Table 3-16. From this table, a conservative exclusion radius value of 10 ft (and the corresponding TDZ value) was assumed for all regions.

TABLE 3-16 . Values of TDZ

Radius	TDZ
5 ft	1.11×10^{-6}
10 ft	9.57×10^{-7}
20 ft	8.01×10^{-7}
30 ft	7.10×10^{-7}
40 ft	6.46×10^{-7}
50 ft	5.96×10^{-7}

Occupational Exposures During Transit

Exposures to truck drivers during transit are calculated using a formula similar to equation (3-7):

$$H = 2 \times T \times (L_i/V_i) \times (K/100) \qquad (3\text{-}12)$$

where H is the occupational exposure in mrem, and the other symbols are as explained above. The term K/100 is in units of mR/hour and denotes the radiation level at the surface of the vehicle. In this case, however, the factor CF is assumed to be 0.195 since the drivers are in the cab of the vehicle.

Occupational Exposures During Stops

Several assumptions must be made concerning the behavior of the drivers during stops. Each stop has been assumed to last one-hour. During this period, it is assumed that the drivers inspect the vehicle for 5 minutes and walk around the vehicle for 10 minutes. The remaining 45 minutes they are assumed to be too far from the vehicle to receive appreciable exposures. Occupational exposures are calculated using the following formula:

$$H = 2 \times T \times S_i \times SD_i \times DF \times (K/100) \qquad (3\text{-}13)$$

where all the symbols have been explained before except DF. This factor is the distance factor accounting for the proximity of the drivers to the vehicle and the shielding afforded by the intervening air. It is explained in detail in reference 5, and based on the above scenario, it can be calculated as follows:

$$DF = (1.0 \times 5/60) + (.907 \times 10/60) = 0.235$$

During the time they inspect the vehicle, they are assumed to be adjacent to the overpack. Consequently, the distance factor is assumed to be unity for 5 minutes. During the time they walk around the vehicle, they are assumed to be about 5 meters from the vehicle. Consequently, the distance factor is 0.907 (reference 5) for 10 minutes. This results in an overall distance factor of 0.235. The correction factor for the finite extent of the vehicle is assumed to be 0.546 in all cases.

3.2.3 Exposures to Waste Handlers During Loading

Occupational exposures resulting from the loading of the waste packages are also included in the transportation occupational exposures. The occupational exposures resulting from waste unloading and emplacement at the disposal facility are considered part of the occupational exposures to the disposal facility personnel. They are considered in Section 4.6, although they are also partially based on the assumptions presented in this section.

The loading occupational exposures are calculated based on two factors: the person-minutes required to load each container, and the overall radiation environment in which loading takes place. The person-minutes shown in Table 3-12 are assumed to be applicable for waste loading. The radiation levels associated with the handling environment (not the package surface radiation levels) are assumed to be the same for each container care level and are given by the index IRE (Table 3-7).

3.2.4 Costs

Transportation costs include a mileage charge (including a fuel surcharge) a cask use charge (rental), and an overweight shipment transportation charge. The mileage charge is calculated by estimating the total shipment miles required (including return trip mileage for casks), using an assumed average distance per one way shipment. In this report, inter and intra-regional shipment distances are assumed to be as given in Table 3-12.

The basic transportation charge depends on the one-way distance, and is assumed according to the following Table 3-17.

TABLE 3-17 . Basic Transportation Charges

One-Way Distance	One-Way ($/mile)	Round-Trip ($/mile)
< 400 miles	2.08	1.54
400-1000 miles	1.81	1.40
> 1000 miles	1.44	1.33

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Added charges, which become significant for extreme-care shipments, include a fuel surcharge (15% of the basic cost) and an overweight charge. The amount of the overweight charge depends on the maximum gross vehicle weight (GVW) allowed in states through which the shipment passes. Any overweight condition up to 85,000 lbs. is charged at $0.26/mile plus the permit charges for each state (about $100 per 600 miles). A GVW of over 85,000 lbs. is additionally charged $0.005 per mile per 100 pounds (cwt) over this limit. For example, for a shipment of 96,000 lbs., (minimum for an extreme-care cask), the charges for a one-way trip of 600 miles would be as follows:

Basic Cost: 1200 miles x $1.40/mile	$1,680.00
Fuel Surcharge @ 15% of base cost	252.00
Overweight Charge: 600 miles x $0.26/mile	156.00
Overweight Surcharge @ $0.005/cwt/mile	330.00
Five Overweight Permits @ $20/state	100.00
Total	$2,518.00
Per Mile:	4.20

The cask use charge calculation assumes an average turnaround time of 4 days. Cask rental rates vary depending on the size and weight of the cask required. They are assumed to average $310/day for shielded casks enclosing high activity waste, and range down to $135/day for an unshielded 120 ft^3 capacity cask. The rental rates also vary with the specific type of nuclear material the cask is licensed to carry and the accompanying performance standards the cask must satisfy to accomodate the various types of nuclear materials. The calculated results for the additional factors are then summed to determine the total transportation cost for the waste. Table 3-18 presents the assumed cask-days between the four regions.

TABLE 3-18 . Cask Days Between Regions

Generating Region	Disposal Region 1	2	3	4
1	2	8	8	20
2	8	3	8	20
3	8	8	5	14
4	20	20	14	8

CHAPTER 3.0 REFERENCES

(1) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Draft Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0782, September 1981.

(2) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Final Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0945, November 1982.

(3) Wild, R.E., et al., "Data Base for Radioactive Waste Management, Volume 2, Waste Source Options Report," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-1759, November 1981.

(4) Oztunali, O.I., et al., "Data Base for Radioactive Waste Management, Volume 3, Impacts Analyses Methodology Report," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-1759, November 1981.

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(6) U.S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, "Final Generic Environmental Statement on Uranium Milling," USNRC Report NUREG-0706, September 1980.

(7) Oztunali, O.I. et al., "Influence of Leach Rates and Other Parameters on Groundwater Migration," Prepared by Dames & Moore for U.S. Nuclear Regulatory Commission, USNRC Report NUREG/CR-3130, September 1982.

(8) U.S. Government Printing Office, "Economic Indicators," Prepared for the Joint Economic Committee by the Council of Economic Advisers, February 1985.

4.0 DISPOSAL IMPACTS

In this report, post-disposal radiological impacts are considered of prime importance in successful performance of a disposal facility, and are thus discussed in greater detail than the operational scenarios. Four sets of generic post-disposal scenarios for human exposure are considered, including:

1. "Intruder scenarios" involving inadvertant human intrusion into the waste;
2. "Groundwater scenarios" involving groundwater migration;
3. "Leachate accumulation scenarios" involving leachate percolation and accumulation within the disposal cells; and
4. "Exposed waste scenarios" in which disposed waste is assumed to be brought to the surface and is dispersed by wind or water.

Other impact scenarios considered in this report include the "operational" scenarios involving impacts during the operation of the disposal facility, disposal facility costs, and land use. Disposal facility costs are considered from the point a decision is made to establish a disposal site to the end of the active institutional control period.

A brief overview of the exposure scenarios considered in this report is presented in Section 4.1. This is followed by Sections 4.2 through 4.6, which discuss the calculational procedures for these scenarios in more detail, and address intrusion, groundwater, leachate accumulation, exposed waste, and operational scenarios, respectively.

4.1 Overview of Impacts Scenarios

As addressed in the following sections, the calculational procedures and assumed parameter values vary for the different exposure scenarios depending upon the following considerations:

- the waste form and packaging characteristics of the waste stream;
- the environment in which the disposal facility is located; and
- the design, operation, and closure of the disposal facility.

As can be seen from the previous sections, the waste form and packaging characteristics of the waste streams are diverse and encompass many of the processing options currently available. This diversity was in fact the reason for utilization of the waste stream specific indices discussed in Section 2.2. These indices permit the analysis of radiological impacts from an almost unlimited number of waste stream combinations.

Likewise, the structure of the codes allows the user to consider an almost unlimited number of disposal facility site environments. In this report, four reference facility site environments have been characterized, and the user of the codes may specify the particular site environment to be considered. Alternatively, the code files may be modified, thus replacing assumed environmental parameter values with site specific ones.

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Similarly, the existing analysis methodology permits the use of six different disposal technologies simultaneously at one facility. These six disposal technologies refer to the distinct classes of waste considered in this report, namely, Class A, Stable Class A, Class B, Class C, Class D1, and Class D2. Any one of these waste classes may be disposed using any one of the distinct disposal technologies parameterized for this report. In addition, site specific variations may be considered by altering the assumed parameter values in the code files.

There are two types of radiological exposure scenarios considered in this report: concentration scenarios (the operational and intruder scenarios) which depend upon the concentrations of the radionuclides within the waste streams considered, and total activity scenarios (the groundwater, leachate accumulation, and exposed waste scenarios) which depend upon the total radionuclide inventory and waste volume disposed at the disposal site (Refs. 1-3). These are briefly discussed in Sections 4.1.1 and 4.1.2, respectively. This is followed by Section 4.1.3 which is a general discussion of some of the decision indices and the manner in which they are used in the impact scenario calculations.

4.1.1 Concentration Scenarios

The concentration scenarios consist of the intruder scenarios and the operational scenarios. Most of the concentration scenarios (the exceptions are the intruder-agriculture scenario and routine operational exposures) are acute exposure events -- i.e., the release of radionuclides and the exposure event takes place over a period much less than a year. The equation generally applicable to these scenarios is:

$$C_a = I \, C_n \tag{4-1}$$

where (C_a) denotes the concentration of radionuclide (n) at the biota access location and (C_n) denotes the radionuclide concentration of the waste, both in units of (Ci/m^3), and (I) is the interaction factor which depends on the specific release scenario considered and is normally dimensionless.

Two major variations of equation 4-1 occur for sealed sources. For these wastes, the variable C_n is given in units of total activity (in Ci) rather than concentration. Consequently, the interaction factor may have units of m^{-2} if C_a is in units of Ci/m^2 and may have units of m^{-3} if C_a is in units of Ci/m^3. These variations are considered individually in the concentration scenarios outlined in Sections 4.2 and 4.6.

The interaction factor (I) can generally be expressed through the following equation:

$$I = f_o \, f_d \, f_w \, f_s \tag{4-2}$$

where all the parameters are dimensionless, and where

f_o = time-delay factor;
f_d = site design and operation factor;
f_w = waste form and package factor; and
f_s = site selection factor.

The time-delay factor (f_o) is expressed as an exponential radionuclide decay factor and incorporates the effects of the time delay between the time that the waste is disposed and the time that the scenario is initiated. This factor is a property of the particular scenario and disposal technology being considered. For the concentration scenarios, it is given by the formula:

$$f_o = \exp[-\lambda\ \mathrm{TDEL}] \tag{4-3}$$

where λ is the radionuclide decay constant in units of year^{-1}, and TDEL is the period (in years) between the disposal of the waste and the initiation of the specific scenario. For the operational scenarios, it is assumed to be zero, which results in a time-delay factor of unity. For the intruder scenarios, it is usually taken equal to the period between the cessation of disposal operations and the assumed loss of institutional controls. As discussed in Section 2.2.1, this period is the sum of the closure period (ICLS), the observation period (IOBS), and the active institutional control period (IINS). However, the value of this factor may also depend on the specific scenario considered (see Section 4.2). Note that, for the intruder scenarios, these assumptions ignore the radioactive decay that would occur during operation of the disposal facility. This practice is conservative, but accounts for the possibility that the inadvertent intruder initiates the scenario at a location containing waste from the last year of disposal facility operation.

The waste design and operation factor (f_d) expresses the fraction of the waste that is available to the transfer agent. It usually depends on the operational scenario and/or the efficiency of the disposal design.

The waste form and package factor (f_w) expresses the resistance of the waste to mobilization by the specific transfer agent initiating the scenario. For example, this factor could be considerably less than unity for waste streams solidified in a matrix and/or packaged in containers that are likely to retain their integrity at the time of specific scenario. This factor is a property of the waste stream as it is being disposed.

The site selection factor (f_s) depends on many parameters, and generally expresses the influence of the disposal site environment on waste release and transfer. For example, for the single container accident scenario and for the inadvertant intruder-construction scenario, it is proportional to the transfer factor between contaminated soil and contaminated air. In some cases, however, it is also proportional to the fraction of a year that the human exposure scenario takes place. Since the pathway dose

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conversion factors presented in Appendix D have been calculated for a full year exposure period, the site selection factor must compensate for this calculational convenience.

4.1.1.1 Intruder Scenarios

In this report, the active institutional control period is assumed to last from the end of the observation period to the point in time when the institutional controls are assumed to break down. After this period, an intruder may unintentionally come across a closed waste disposal site, and subsequently modify it for a specific purpose, such as housing construction or agriculture. As a result, short- and long-term radiation exposures to the individual can ensue. Four intruder scenarios are considered in which these exposures are calculated.

Three of the intruder scenarios (intruder-drilling, intruder-construction, and intruder-discovery) are acute exposure events. That is, the exposure occurs for a limited period of time (less than a year). The intruder-agriculture scenario, however, is assumed to be chronic, since it is possible (but very unlikely) that the intruder would live for several years at the site before it is discovered that there is a hazard. There are two minor variations of the intruder-agriculture scenario depending on the event initiating the scenario. The initiating event can be either the intruder-construction scenario or the intruder-drilling scenario.

A brief description of the four intrusion scenarios is presented below. Specific values of the transfer factors used to calculate impacts are discussed in Section 4.2.

The intruder-drilling scenario assumes that an inadvertant intruder drills into the waste for the purpose of establishing a water well for a house to be built. During the drilling activities, some waste is brought to the surface mingled with soil and is contained in a mud pit used by the drillers. The drilling contractor, who is not the same person as that initiating the following intruder-construction and intruder-agriculture scenarios, is then exposed to a source of direct gamma radiation from the waste/soil/water mixture contained in the mud pit. After construction of the well, the drill cuttings are assumed to contribute to impacts from the following intruder-agriculture scenario. (A more complete discussion of drilling methods and options is included in Appendix E.) It would be unreasonable to expect the inadvertant intruder to initiate housing construction at a comparatively isolated location before assuring that water for home and garden use will be available. Thus, this scenario is assumed to precede the following three scenarios.

The intruder-construction scenario involves an inadvertant intruder who may choose to excavate or construct a building on a disposal site. Under these circumstances, dust will be generated from the application of mechanical forces to the surface materials (soil, rock) through tools and implements (wheels, blades) that pulverize and abrade these materials. The dust particles generated are entrained by localized turbulent air currents. These suspended particles can thus become available for inhalation

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by the intruder. The intruder may also be exposed to direct gamma radiation resulting from airborne particulates and by working directly in the waste/soil mixture. For convenience, this scenario is called the intruder-construction scenario, and appropriate parameter values appliable to typical construction activities are used.

The intruder-discovery scenario is similar to the intruder-construction scenario except that the intruder is able to recognize that he is digging into very unusual soil immediately upon encountering the wastes, and consequently, gets exposed for a much smaller period. This scenario is applicable for stable waste forms which are disposed in a segregated manner from unstable waste forms. These stable waste forms are assumed to be recognizable as something out of the ordinary for a period of several hundred years after disposal.

The intruder-agriculture scenario involves an inadvertant intruder who is assumed to occupy a dwelling located on the disposal facility and ingest food grown in contaminated soil. The soil is assumed to be contaminated as a result of spreading the excess dirt dug up during the intruder-construction scenario. Garden crops may be subject to radionuclide contamination as a result of direct foliar deposition of fallout particulates. Garden crops may also uptake radionuclides via soil-to-root transfer from contaminated soil. The soil may be initially contaminated, or it may become contaminated as a result of deposition. The inadvertant intruder may also be exposed to direct ionizing radiation such as beta and gamma radiation from the naturally suspended radioactivity and from the waste/soil mixture. He also may inhale contaminated air particulates. (See Appendix D for the uptake pathways considered.) This scenario is called the intruder-agriculture scenario, and can occur after the intruder-drilling and the intruder- construction scenarios. It would be precluded by an intruder-discovery scenario.

4.1.1.2 Operational Scenarios

During the disposal facility operational life, there will be routine as well as nonroutine exposures to facility workers. Routine exposures are chronic exposure events -- i.e, they occur during most of the time, year after year -- while nonroutine exposures are acute exposure events -- i.e., they are hypothetical and would occur for only a fraction of a year.

Three types of routine exposures associated with disposal facility operation are considered in this report. The first type depends primarily on the annual number of the transportation vehicles arriving at the site, and involves exposures associated with checking in, inspection, decontamination, and checking out the vehicles. The second type of routine exposures depends primarily on the waste packaging and shipping mode parameters and secondarily on the disposal technology used, and involves the unloading and emplacement of the waste into the prepared disposal cells. The third type of routine exposures involves routine maintenance of the disposal cells. These scenarios are discussed in Section 4.6.3 and Appendix C.

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Finally, there may occur operational accidents which result in nonroutine exposures to the disposal facility workers and/or the population surrounding the facility. Two operational accidents are considered in this report. The first accident involves a single container which is dropped during handling, breaks open, and results in dispersal of some of the radioactivity. The second accident involves a fire in one of the disposal cells during which the radioactivity is entrained in the smoke and is carried offsite. These two scenarios are discussed in Section 4.6.

4.1.2 Total Activity Scenarios

These scenarios depend upon the entire activity disposed at the site. All of the total activity scenarios are chronic exposure scenarios (i.e., continuous release and exposure). The equation applicable to the total activity scenarios for each radionuclide is:

$$C_a = \sum_i I_{in} C_{in} \qquad (4\text{-}4)$$

where (C_a) and (C_{in}) denote the concentrations of radionuclide (n) at the biota access location and in the $(i)^{th}$ waste stream, respectively, in units of (Ci/m^3), and (I_{in}) is the radionuclide-specific interaction factor between the $(i)^{th}$ waste stream and the biota access location. The capital sigma indicates that the total radionuclide concentration at the biota access location is a summation of the radioactivity contributed by each waste stream. This summation may also include any potential integration that must be performed due to the areal extent of the disposal site and the areal distribution of the waste streams.

In a manner similar to the concentration scenarios, a major variation of equation 4-4 occurs for sealed sources. For these wastes, the variable C_{in} is given in units of total activity (in Ci) rather than concentration. Consequently, the interaction factor has units of m^{-3}. This variation is considered individually in the total activity scenarios outlined in Sections 4.3 through 4.5.

The interaction factor I_i (dropping the radionuclide subscript for simplicity) can generally be expressed through the following equation:

$$I_i = f_o f_{di} f_{wi} f_{si} \qquad (4\text{-}5)$$

where:

f_o = time-delay factor (dimensionless);
f_{di} = site design and operation factor (dimensionless);
f_{wi} = waste form and package factor (m^3/yr); and
f_{si} = site selection factor (yr/m^3);

and where the subscript i denotes the waste stream. The factor f_o is given by equation 4-3, and the values of f_{di}, f_{wi}, and f_{si} may be functions

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of the properties of individual waste streams. Total activity scenarios include the groundwater scenarios, the leachate accumulation scenarios, and the exposed waste scenarios.

Groundwater Scenarios

There are four groundwater scenarios depending on the assumed biota access location. One of the access locations is an on-site well which may be drilled into an unconfined aquifer beneath the site and used by a potential inadvertant intruder (intruder well scenario). The second access location is a well drilled into the unconfined aquifer at the boundary of the site just beyond the buffer zone (boundary well scenario). A third access location is a well drilled into the unconfined aquifer some distance away from the disposal facility and pumped for common use by a small population (population-well scenario). The fourth location is a stream that receives the discharge from the unconfined aquifer and which may be used by a larger population (surface water scenario).

In this report, it is assumed that the water table gradient underneath the disposal site is unidirectional, and that the intruder-well is located at the downgradient boundary of the disposal area. This location is more conservative than a well located in the middle of the site; only about half of the potential effluent from the site would contribute to the contamination at a well in the middle of the site whereas all of the potential effluent would contribute to a well at the assumed location. These scenarios are discussed in Section 4.3.

Leachate Accumulation Scenarios

These scenarios are conceptually associated with the groundwater scenarios and are initiated by rainwater percolation into and accumulation within the disposal cells. There are three distinct leachate accumulation scenarios discussed in Section 4.4:

(1) During operation of the disposal facility, accumulating leachate is continuously removed by site operators, processed in some manner, and released to a nearby stream.
(2) Accumulating leachate is allowed to completely fill up the disposal cells to the point that the leachate overflows and enters a nearby stream.
(3) Rather than overflow, the accumulated leachate in the second scenario above is assumed to be removed and processed through an evaporator, and a fraction of the activity is released into the atmosphere.

Exposed Waste Scenarios

In these scenarios, part or all of the surface area of the disposed waste is assumed to be exposed through some means, and this exposed waste is assumed to be accessed by transfer agents such as wind or water. The mechanism that initiates uncovering of the waste may be an intruder scenario as discussed previously, or it may be erosion of the waste cover by surface water or wind action. Initiating mechanisms related to human

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intrusion activities are examined in Section 4.2 while initiating mechanisms related to erosion of the waste cover are examined in other references (Refs. 3, 4).

There are two basic types of exposed waste scenarios depending on whether the transfer agent is wind or surface water. For the wind transport scenario, only population exposures are considered; individual exposures are bounded by the above intruder-construction and intruder-agriculture scenarios. The entire exposed waste area is assumed to be a point source for the impact calculations since the population is assumed to be comparatively distant. For the surface water transport scenarios, exposures to individuals consuming water from an open water access location are considered. Again the disposal facility is considered a point source for this scenario. The equations used in the calculations are examined in Section 4.5.

4.1.3 Decision Indices and Impact Scenarios

The decision indices were defined in Section 2.2, and a discussion of the manner in which they are used in the impact scenarios was deferred to this chapter. The use of most of the various decision indices discussed in Section 2.2 are either self-evident or they do not affect impact calculations. For example, the general facility indices, IR, IDAT, IOFL, IBUF, NBRN, and NBES, and the schedule indices, IBEG, IEND, ICLS, IOBS, and IINS have clear and obvious uses. In addition, some of the waste stream specific indices, e.g., I1, I2, I3, IRI, IBLG, NDXS, and NDST, also have unambiguous and well defined uses. Consequently, only two categories of decision indices remain to be considered: the waste form and behavior indices, I4 through I10; and the disposal technology indices, IU, ID, IT, IC, IE, IB, IX, IS, and IM. This section considers the waste form and behavior indices. The disposal technology indices are discussed in the subsequent scenario sections, i.e., Sections 4.2 through 4.6.

Accident Index - I4

The accident index (I4) is composed of two single-digit subindices which relate to the potential for waste forms to disperse into the atmosphere as a result of two types of hypothetical site operational accidents. The first subindex (ISC) is termed the scatter index while the second subindex (IFL) is termed the flammability index.

The scatter subindex (ISC) is a measure of the potential for the contents of a waste container to be dispersed into the air due to an operational accident in which the waste container is severely damaged. This could hypothetically occur from an accident in which the waste container is dropped from a significant height such as from a crane. This index is primarily used in the single-container accident scenario (see Section 4.6.1). Waste forms which are assumed to have a low probability of becoming suspended into respirable particles are assigned an ISC index value of 3. Those waste forms which are assumed to have a high probability of becoming suspended into air (such as incinerator ash) are assigned an ISC index value of 0. Waste forms which tend to crumble or fracture easily

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are assigned an index value of 1. Waste forms consisting of a mixture of material with ISC indices of 1 and 3 are assigned an index value of 2.

The scatter index is associated with a multiplier, denoted by f_C and used to modify the waste form and package factor f_w, which designates the fraction of the waste released into the air from this accident. Assumed values for the accident scatter index and associated multipliers for various typical waste forms are given below.

ISC	Multiplier f_C	Example Waste Forms
0	1.0	Dewatered sludge, ash, dirt, and other miscellaneous powders
1	0.1	Trash, dewatered resins
2	0.01	Waste solidified in cement
3	0.001	Waste solidified using vinyl ester styrene, sealed sources

In other words, the multiplier is generally given by the relationship 10^{-ISC}. However, the multiplier is assumed to be zero for for activated metals and also zero for sealed sources which are contained in a high integrity container. The above multiplier is primarily used for comparison purposes.

In a similar manner, the flammability subindex (IFL) is a measure of the potential for the waste to disperse into the air in the event of an accidental operational fire. The only scenario in which this index is utilized is the accident-fire scenario (see Section 4.6.2). Each waste stream is subjected to the accident-fire scenario separately. The scenario is assumed to be possible only if (1) the waste stream being tested can support combustion, or (2) the waste stream being tested is mixed during disposal with other waste streams containing combustible material.

This index is also associated with a multiplier, f_F, which designates the fraction of the waste released into the air from a fire and is used to modify the waste form and package factor, f_w. In one study (Ref. 5), the fraction of waste released into the atmosphere as the result of an accidental fire involving waste is estimated to be about 10^{-2} for combustible material and about 10^{-5} for unsolidified resins; it is estimated in another study (Ref. 6) that the fraction of combustible material released from an accidental fire involving waste is about 10^{-3}. In another study (Ref. 3), a radionuclide specific release fraction is assumed based upon EPA measurements and estimates of particulate release from open burning of combustible municipal waste. This combustion process is very inefficient (similar to an operational accidental fire), and leads to a calculated release fraction for non-volatile radionuclides of 0.019. In the study, a release fraction of 0.038 was also assumed for all volatile and semi-volatile radionuclides except H-3 and C-14. For these nuclides release fractions equal to 0.90 and 0.75, respectively, were assumed.

This report uses a combination of the calculational methodologies used in References 2 and 3. Radionuclide-specific release fractions are assumed for waste streams, and then a multiplier is applied which relates the relative flammability of different waste forms. The radionuclide-specific release fractions, f_r, obtained from Reference 3 for flammable waste streams are as follows:

Radionuclide	Multiplier f_r
H-3	0.90
C-14	0.75
Tc-99	0.038
Ru-106	0.038
I-129	0.038
Particulates	0.019

In this report, waste streams with flammability indices of 0 and 3 are assumed to have a waste form multiplier of 1.0 and 0.000125, respectively. The waste streams with flammability indices between these two extremes have been assigned a waste form multiplier calculated from the geometric mid-points of these two values (each index value is 20 times the adjacent lower index value). The following table gives the assumed waste form multipliers for the respective indices.

IFL	Waste Form Multiplier
0	1.0
1	0.05
2	0.0025
3	0.000125

In other words, f_F is the product of f_r and 20^{-IFL}. These assumptions are extremely conservative. The release fraction for combustible material is assumed to a factor 10 to 100 higher than in other studies (Ref. 5, 6). The assumed fraction for non-combustible material (IFL = 0) is slightly greater than the value previously quoted for unsolidified resins. The flammability indices assigned to the waste streams considered in this study are presented in Appendix B.

Dispersibility Index - I5

This index is primarily used in scenarios that involve airborne dispersion of radioactivity, and is a measure of the relative dispersibility of the waste form, in comparison with soil, a long time after disposal. Assigning relative dispersibilities to waste streams a long time after disposal is speculative; however, the resistance of the waste form to chemical and biological attack can be used as a guide to determine this property. Another property of the waste form that can be used to estimate the comparative values of this property is the compressive strengths of the waste forms. These are discussed in References 1 and 2.

This index triggers the use of a correction factor, f_D, applied to the waste form and package factor fw. As an upper bound for this property, the most dispersible waste form has been assumed to be equivalent to soil, and no credit has been considered due to waste form. These forms have been assigned an index value of 0, and an f_D value of 1. This value is believed to be conservative in view of the fact that the fraction dispersible into respirable particles of powder PuO_2 packages in transportation accidents has been assumed in the past to be 0.001 (Ref. 7).

In comparison, waste forms such as trash are taken to be not as readily dispersible into respirable particles as waste streams such as filter sludges. These wastes easily decompose; however, the decomposition residues are likely to contain water and other liquid decomposition products which will still cause the residues to aggregate into a less dispersible state. Similarly, unsolidified ion exchange resins would appear to be less dispersible into respirable particulates than filter sludges.

Based on discussion and data presented in Appendix B, waste forms solidified using solidification scenarios A have been assigned an ID value of either 1 or 2 and an f_D factor ranging from 0.1 to 0.01. Waste streams solidified using solidification scenario C have been assigned an ID value of either 2 or 3 and an f_D factor ranging from 0.01 to 0.001. Thus, in summary the the factor f_D is given by the relationship 10^{-I5}.

The primary use of ID index in the original Part 61 analysis methodology was to illustrate the comparative confinement potential that could result from improved waste form with regard to dispersibility. It still can be used to perform sensitivity analyses on the additional benefits that can result from improved dispersibilities. It should be reiterated, however, that this index is speculative. Consequently, a general facility index (NBES - see Section 2.2.1) has been formulated that may optionally be used to cancel the effects of this index.

Leachability Index - I6

As stated previously, this index is a measure of a waste form's resistance to leaching and is primarily determined by the solidification procedures used. Its primary purpose is to assign values to the estimated leachability potential of solidified waste streams in comparison with unsolidified waste streams. As discussed in Reference 8, radionuclide-specific leaching fractions for unsolidified waste streams have been estimated to the extent possible based upon actual leaching data from existing disposal facilities. The leachability index assigns values to a multiplier of these unsolidified waste stream leaching fractions. The product of the multiplier and the unsolidified waste leaching fraction gives, for each waste stream, the actual leaching fractions used in the impact calculations. The multiplier is assigned a value of unity for unsolidified waste streams such as trash or dewatered resins, and a value less than unity for solidified waste streams. The multiplier value assigned to solidified waste streams is dependent upon the particular solidification scenario considered.

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In this report, four different solidification scenarios are considered. The first scenario, corresponding to an I6 value of 1, indicates that no credit for improved leaching capability has been considered. A level of waste form performance corresponding to an I6 value of 2 (also called solidification scenario A) has been simulated by assuming that the waste is solidified using cement. A level of waste form performance corresponding to an I6 value of 3 (also called solidification scenario B) has been simulated by assuming that the waste is solidified using synthetic organic polymers. Finally, a level of waste form performance corresponding to an I6 value of 4 (also called solidification scenario C) has been simulated by assuming that the waste is solidified using synthetic organic polymers of the highest quality and implementing the state-of-the-art in solidification procedures. (These scenarios are also referred to as the base case, cement case, synthetic polymer case, and optimistic case, respectively.)

Based on the data presented in Reference 8, the following multipliers have been assigned to the various leachability indices in this report.

Scenario	I6	Multiplier
Base	1	1
Cement	2	1/30
Synthetic Polymer	3	1/160
Optimistic	4	1/400

These values are applicable primarily to the groundwater scenarios. Another scenario which is affected is the food (soil) uptake pathway of the intruder-agriculture scenario since the level of contamination in interstitial soil water available to vegetation may depend on the leachability of the waste. The use of the leachability index in the intruder-agriculture and groundwater scenarios is discussed in Sections 4.2 and 4.3, respectively. The values assigned to the I6 index, however, may be modified further depending on the properties of the waste and the disposal technology implemented (see below).

It should be pointed out that this index is used in a manner similar to the dispersibility index above. It is speculative and the credits assumed for various solidification scenarios cannot be totally justified based on the available data. Consequently, in a manner similar to the dispersibility index, the general facility index (NBES) may optionally be used to cancel the effects of this index as well. However, as alluded to above, the index can be used as a sensitivity analysis tool to assess the relative benefits that can result from improved waste forms.

Chemical Content Index - I7

As stated, this index denotes whether a waste stream may contain chelating or organic chemicals. An index value of 0 indicates the likelihood that these agents are absent in the stream, whereas an index value of 1 indicates that the stream is likely to contain chelating or organic chemicals.

This index, in conjunction with the segregation option index IS, is used to modify the multiplier values assigned to the leachability indices for the groundwater and intruder-agriculture scenarios. This multiplier is denoted by the symbol f_l, and the following table is used in determining the fraction leached from a particular waste form:

I6	IS=1 and I7=0	IS=0 or I7=1
1	1	1
2	1/30	1/7.5
3	1/160	1/40
4	1/400	1/100

This table should be interpreted as follows. For a specific waste stream with a given leachability index (I6), if the waste stream either contains chelating agents (I7=1) or is disposed mixed with other waste streams containing chelating agents (IS=0), then the higher leach fraction multiplier is used. If the waste stream does not contain chelating agents (I7=0) and it is not mixed with other wastes containing chelating agents (IS=1), then the lower leach fraction multiplier is used.

A similar procedure is applied to the soil retardation coefficients assigned to individual radionuclides. Retardation coefficients denote the potential of the disposal facility site soils to retard the radionuclides during groundwater migration. If there is no waste segregation at the disposal facility, then the retardation potential of the disposal site soils is assumed to be reduced as discussed in Section 4.3.

Stability Index - I8

This index primarily denotes whether the waste form is likely to reduce in volume after disposal due to compressibility, large internal void volume, and/or chemical and biological attack. A secondary function of this index is to specify the stabilization technique used during waste processing. This index is primarily used in conjunction with the disposal technology indices IC and IX to determine the amount of percolating precipitation entering the disposal cell. This is explained later in this chapter in Section 4.3.

Activated Metal Index - I9

This index denotes whether the waste stream is an activated metal waste stream (an index value greater than 0) or whether it is not (an index value of 0). An index value of 1 or higher is used to access the following information concerning the activated metal waste stream: (1) the corrosion time in years, i.e., the period of time after which the waste stream can be assumed to have corroded completely under fully saturated conditions, (2) the air dispersibility of the waste stream, (3) the water solubility of the waste stream, and (4) the self-shielding afforded against direct radiation by the waste stream. These four values are waste stream specific, and are input from a data file called METALS.DAT. They are used as follows.

The first property is predicated on the fact that each distinct waste stream, such as a PWR core shroud sectioned in a specific geometry or a BWR reactor vessel sectioned in another specific geometry, will take a different period to corrode entirely. This time period depends on the corrosion penetration per year which depends on the material composition of the waste (e.g., carbon steel, 304 stainless steel), and on the surface to volume ratio of the activated material which depends on its geometry. This property is called the corrosion time, and denoted by the symbol f_{AC} in years. It is used as a delay factor for the intruder-drilling scenario (see Section 4.2). A corollary property to the corrosion time is the corrosion fraction per year. This corrosion fraction (inverse of f_{AC}) is used in several intruder scenarios to modify the waste form and package factor, f_w, as explained below.

The second and third properties depend on the fraction corroded per year, the material properties of the waste, and the chemistry of the corrosion products with regard to the transfer agents. Each distinct waste stream is assigned two accessibility multipliers, f_{AD} and f_{AL}, which are used to multiply the transfer coefficients assumed for routine unsolidified waste streams. The factor f_{AD} replaces the dispersibility multiplier, f_D, for these waste streams, and the factor f_{AL} is used to multiply the M_o fractions used in groundwater scenarios.

In this report both f_{AD} and f_{AL} are assumed to be equal to the corrosion fraction per year (inverse of f_{AC}) multiplied by the time at which the scenario occurs up to a maximum value of unity. No special credit is assumed for the dispersibility or the solubility of the corrosion products. Only the noncorroded fraction of the waste stream is exempted from the applicable scenario. However, if data can be found on these properties of the corrosion products, these multipliers can be altered accordingly by modifying the values in METALS.DAT.

The fourth property, self-shielding, is a function of the geometry of the waste and its material properties. It is denoted by the symbol f_{AG}, and is calculated for each waste stream and is increased linearly to equal unity when the material is completely corroded ($f_{AG} = 1$ at f_{AC} years).

Source Index - I10

This index denotes whether the waste stream is not a source waste stream (an index value of 0) or whether it is (an index value greater than 0). Source waste streams are treated in a special manner in this report.

First of all, radioactive concentrations are less meaningful than the average total activity of a source waste stream. They are extremely small (usually no more than 1" in diameter and 6" long) and may contain a significant amount of activity. Thus, the variable C_n in equation 4-1 and C_{in} in equation 4-4 are given as total activities in units of Ci/source. Second, in a manner analogous to the activated metals, their interaction

C16

with the transfer agents are treated on a waste stream specific basis. Thus, an I10 index value of 1 or higher is used to access the following information concerning the source waste stream: (1) the delay time applicable to the waste stream for intrusion scenarios, (2) the air dispersibility of the waste stream, (3) the water solubility of the waste stream, and (4) the self-shielding afforded against direct radiation by the waste stream. These four values are waste stream specific, and are input from a data file called SOURCE.DAT. They are used as follows.

The first property is predicated on the fact that each distinct waste stream may have a different isolation measure applied to it. For example, some sealed sources may be encapsulated in specially fabricated high integrity containers (HIC). In this case, the waste stream is assumed to be inaccessible before the complete corrosion of the HIC. This delay time is denoted by f_{SC}, and is used in the inadvertant intruder scenarios (see Section 4.2).

The second and third properties depend on the material properties of the waste and its chemistry with regard to the transfer agents. Each distinct waste stream is assigned two multipliers, f_{SD} and f_{SL}, which are used to multiply the transfer coefficients assumed for routine unsolidified waste streams. The factor f_{SD} replaces the dispersibility multiplier, f_D, for these waste streams, and the factor f_{SL} is used to multiply the M_o fractions used in groundwater scenarios. In this report both f_{SD} and f_{SL} are assumed to be equal to unity, i.e., no special credit is assumed for the dispersibility or the solubility of source waste streams. However, if data can be found on these properties of the source waste streams, these multipliers can be altered accordingly by modifying the values in SOURCE.DAT.

The fourth property, self-shielding, is a function of the geometry of the waste and its material properties. It is denoted by the symbol f_{SG}, and is calculated for each waste stream and is increased linearly to unity as a function of time to account for the reduction, if any, in shielding with the passage of time.

Other properties associated with source waste streams are its volume, and the maximum cross sectional area presented by each source. Primarily due to a lack of data, these two properties will be assumed to be independent of the waste stream. The volume, V_S, will be assumed to equal 309 cm^3 (a 6" long cylinder is assumed with a radius of 1"), while the maximum cross sectional area, S_S, will be assumed to equal 78 cm^2 (2" by 6"). These two values are used in the groundwater scenarios (see Section 4.3).

4.2 Intruder Scenarios

The intruder scenarios include those involving direct inadvertant human contact with the disposed waste. In these scenarios, potential exposures are calculated considering only the radionuclide concentrations in the waste streams assumed to be actually contacted by the intruder. The radionuclide concentrations in the part of the disposal facility not contacted by the potential inadvertent intruder do not enter into the calculations.

These concepts are further expanded in the following four sections which present the calculational procedures for determining exposures from the four basic intruder scenarios considered in this report. These include the intruder-drilling scenario presented in Section 4.2.1, the intruder-construction scenario presented in Section 4.2.2, the intruder-discovery scenario presented in Section 4.2.3, and the intruder-agriculture scenario presented in Section 4.2.4. Finally, the effects of radionuclide chains and daughter ingrowth on the various intruder scenarios are considered in Section 4.2.5.

Each scenario section initially presents the assumptions and formulations for routine wastes, i.e., those that are neither activated metals (I9=0) nor source waste streams (I10=0). This is followed by a consideration of the differences and variations resulting from the activated metal (I9>0) and source (I10>0) waste streams.

4.2.1 Intruder-Drilling Scenario

It is assumed in this scenario that, some time after the end of operations at the disposal facility, institutional controls temporarily break down and an intruder inadvertently decides to have a house built on the disposal facility. In order to do so, however, he must first install a well to secure an adequate supply of water for his living needs. He contracts a local well drilling company to have the well installed, and the company sends a crew of two persons equipped with a hydraulic (or mud rotary) drilling rig. The crew inadvertantly drills through the waste bringing up a small amount of waste/soil mixture mixed with the drill cuttings and mud. The geometry of the drilling operation is shown in Figure 4.1.

As shown, the cuttings are contained in a "mud pit." The cuttings from the drill hole settle out within this pit, and relatively clear drilling mud overflows into the next pit and is recirculated. The crew is exposed to direct gamma radiation from the waste contained in this mud pit. Possible inhalation impacts are discounted given the liquified nature of the contaminated mud. The following are the basic assumptions of this scenario:

(1) Drilling is assumed to be performed using a hydraulic rotary rig, which is a commonly utilized type of drilling equipment. It is relatively inexpensive and mobile, and can be used in almost any site environment. Use of cable-tool rigs, which are cheaper and just as mobile, is another popular method of drilling; however, they are much slower than rotary rigs. They have largely been replaced by rotary rigs in some parts of the U.S. (see Appendix E and Ref. 9).

(2) If resistance is encountered during the drilling (as would be the case if the drill bit hit solid rocks or intact metal), the crew would simply move a few yards horizontally and drill a new hole. Drilling equipment and bits that can drill through solid rock and intact metal are available; however, they are more expensive than the equipment and bits that are normally required and used for water well installation. Similarly, it is anticipated that standard drill bits would have difficulty penetrating reinforced concrete

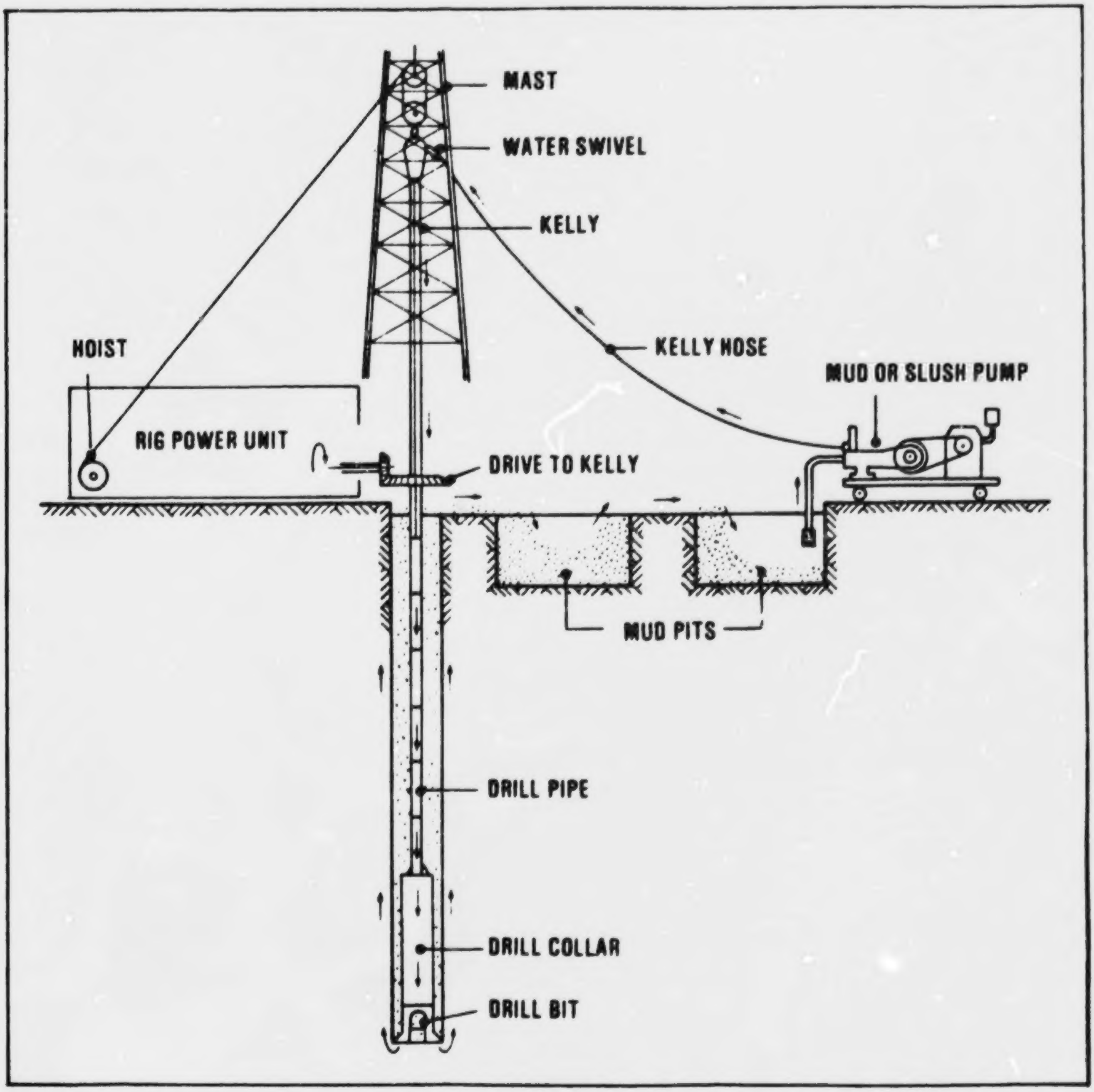

MAST
WATER SWIVEL
KELLY
HOIST
KELLY HOSE
MUD OR SLUSH PUMP
RIG POWER UNIT
DRIVE TO KELLY
MUD PITS
DRILL PIPE
DRILL COLLAR
DRILL BIT

Figure 4.1 Representation of Drilling Scenario

structures, although little difficulty is expected for waste stabilized using grout backfill.

(3) The diameter of the drill hole is assumed to be 8 inches, which would permit the installation of an 4 inch diameter casing for the well. This well provides an adequate supply of water for an average rural family of six.

(4) The hole is assumed to be drilled to a depth of 200 ft. Thus, the volume of cuttings from the hole is calculated to be about 1.98 m^3. In order to assure an adequate supply of water, the well screen must be set some distance below the water table). Consequently, the drilled hole is assumed to be capable of penetrating all the near-surface disposal technologies considered in this report.

(5) Well drilling is only a portion of the well installation process; it usually contains five distinct steps consisting of drilling, casing, installation of the screen, grouting, and developing and pump installation (some of these steps can be performed simultaneously). Drilling takes about 3-4 hours. After the drilling of the hole, the same crew is assumed to complete the well by performing the rest of the above steps. The total time assumed for the activity is 6 hrs.

(6) The mud pit dimensions are assumed to be 8' by 9' by 4' deep. This pit has a volume of about 8.2 m^3 of which 6.1 m^3 is filled with drilling fluid (i.e., the pit is filled with drilling fluid up to about 1 ft below the top of the pit). It is recommended that the volume of the mud pit be approximately three times the volume of the cuttings anticipated from the drilling operation (Refs. 9, 10).

(7) The cuttings from the drilling operation rapidly settle out in the mud pit and could conceivably occupy the bottom 1 ft (there will be less cuttings at the beginning of operation - there may be up to 1 ft at the end). Thus, on the average, the radiation emitted by the cuttings is shielded by a 2-3 ft thick water layer. Moreover, the waste will be encountered relatively early during the drilling, and waste cuttings will be shielded by other cuttings. It is assumed in this report that this shielding water layer is 2 ft thick.

(8) The most exposed individual during this scenario is the driller's helper who stands adjacent to the mud pit.

(9) After use, the mud pit is filled with soil.

4.2.1.1 Routine Wastes

The equation used to calculate individual exposures is as follows:

$$H = \sum_{n} I_{DG} C_n \, \text{PDCF-5} \qquad (4\text{-}6)$$

where H is the 50-year dose committment in mrem, the radionuclide is denoted by n, I_{DG} is the interaction factor given by equation 4-2 (the subscript DG stands for the direct-gamma scenario), C_n is the concentration in the waste, and PDCF-5 is the radionuclide specific pathway dose conversion factor discussed in Appendix D. The interaction factor I_{DG} is given by the following:

$$I_{DG} = 1.12 \times 10^{-5} \text{ CF SF DTK EMP } f_{on} \qquad (4\text{-}7)$$

where the factor 1.12×10^{-5} (in m^{-1}) is the product of the area of the drill hole (324 cm^2) divided by the dilution volume (1.98 m^3) and multiplied by the exposure duration factor corresponding to 6 working hours per year ($6.85 \times 10^{-4} = 6/8760$).

The factor CF is the areal exposure factor. It is approximated using the equations developed for the factor CF in Section 3.1 of Reference 3 for waste transport personnel. In the case of concern, the individual is exposed while standing adjacent to an area of 6.69 m^2 (8'x9'). A diagram of the exposure geometry is given in Figure 4.2.

As shown, the smallest maximum distance from the individual to an element of the exposure source is 8'. The CF value is calculated by conservatively deforming the area of exposure into a pie-slice shaped segment of a disk source whose radius is 8'. The CF value for the disk source of radius 8' is calculated to be 0.234. This value is then prorated to an area of 6.69 m^2 using the area of the disk source of 8' radius (18.68 m^2), yielding an approximate value of 0.0838.

The factor SF is the shielding factor for water 2 ft thick. It is radionuclide specific and is calculated using the following equation:

$$SF = \exp[-u_w t]\, B(u_w t) \qquad (4\text{-}8)$$

where u_w is the linear attenuation coefficient of water (m^{-1}), t is the thickness of the shielding layer (m), and where B(x) is the Berger buildup factor given in Section 3.2.2. The SF factor is calculated using the average energy per gamma emitted for each radionuclide, and the energy dependent linear attenuation coefficients for water (Ref. 11). For example, for Cs-137, using the linear attenuation coefficient for water of 8.62 m^{-1} for a gamma energy of 662 keV, SF is calculated to be 0.0818.

The variables DTK (in meters) and EMP (dimensionless) are the disposal technology specific variables discussed in Section 2.2.3. The factor f_{on} is given by equation 4-3. Combining these values yields

$$I_{DG} = 9.39 \times 10^{-7} \text{ SF DTK EMP } f_{on} \qquad (4\text{-}9)$$

There are several additional considerations applicable to this scenario. One consideration concerns the cuttings suspended in the drilling fluid above the settled cuttings in the mud pit. These could theoretically result in some exposures, although it is difficult to estimate the fraction of cuttings that could remain suspended. A compensating factor is that only 2 ft of shielding is assumed, while the actual shielding thickness could range up to 3 ft.

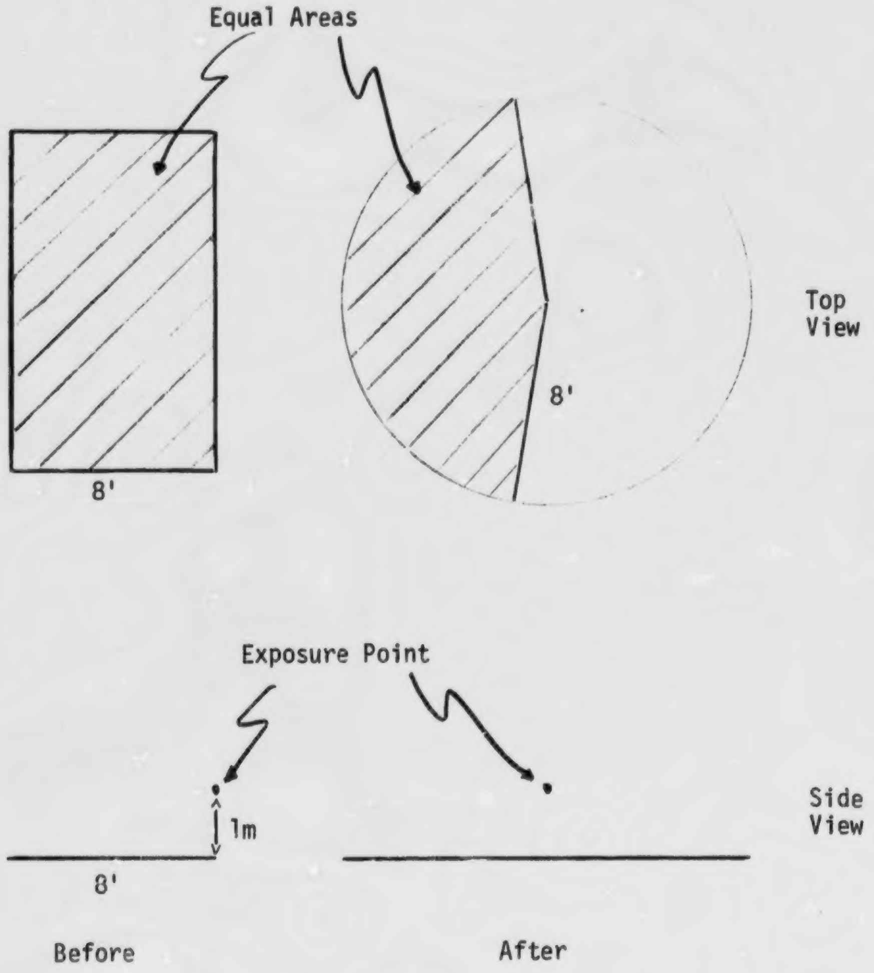

Equal Areas
8'
8'
Top View
Exposure Point
1m
8'
Side View
Before
After

Figure 4.2. Drilling Scenario Exposure Geometry

Another consideration involves the exposure schedule of the scenario. It is assumed that the entire exposure period of 6 hours occurs after the entire waste/soil mixture is drilled through. It would actually take some time to reach and drill through the waste/soil mixture. This factor is conservatively neglected in this scenario.

In addition to the assumed 3-4 hour duration of the drilling, an exposure period of two hours has been assumed to account for the emplacement of the well casing and screen, grouting and development of the well, and the installation of the pump. During this period, it is difficult to estimate an occupancy factor for the most exposed individual. Thus, it is assumed that the exposures occur at the levels experienced during drilling for the entire six hours.

Another consideration concerns the location of the well. It is assumed that the well is drilled at some distance from the planned location of the house. There are legal and regulatory, as well as common sense, restrictions on the proximity of wells drilled for home use to septic fields, building foundations, and sewer lines.

After the completion of the well, the mud pit is assumed to be filled in with the soil excavated to establish the mud pit (some of this soil will also probably be used to fill the space between the borehole and the casing). Subsequently, housing construction begins, during which considerable excavation and grading activities occur. The drill cuttings entombed in the mud pit are assumed to contribute to a subsequent intruder-agriculture scenario unless the intruder-discovery scenario occurs, i.e., the intruder discovers that the material underneath is very unusual and investigates prior to proceeding further.

A final consideration is application of the drilling scenario for waste disposed in disposal methods involving extensive use of reinforced concrete (ID = 10 through 17). It is assumed that drilling through reinforced concrete would be sufficiently difficult, or sufficiently out of the ordinary, that the drilling crew would stop and shift to a different drilling location. No, or comparatively little, impacts from the drilling scenario would occur.

This assumption is assumed to be applicable for only so long as the reinforced concrete structure can be assumed to be structurally stable. This limiting time period, as in the following Section 4.2.3 for the intruder discovery scenario, is assumed to be 500 years following the end of the surveillance period. Following this 500 year period, the scenario is assumed to be fully applicable. (The scenario is assumed to be fully applicable for grouted waste at any time.)

4.2.1.2 Activated Metal Wastes

The interaction factor I_{DG} for activated metal waste streams is also given by equation 4-9 except for the factor f_{op}. This factor is calculated by equation 4-3 provided that TDEL exceeds the time it takes for the metal to corrode completely, i.e, f_{AC} as discussed in Section 4.1.3. If f_{AC} exceeds

TDEL, however, f_{on} is taken to be zero. This corresponds to the assumption that drilling would not occur through intact metal.

4.2.1.3 Source Waste Streams

For source waste streams, the interaction factor is considerably different. It is assumed in this report that no more than one source may be accessed during the intruder-drilling scenario. This is predicated on the assumption that each source will be packaged in its own container and disposed separately (see Appendix E). As stated above (see Section 2.2.3), the variable C_n in equation 4-6 for source waste streams stands for the average total activity for each source. Consequently, the interaction factor I_{DG} has the dimensions of m^{-3}. It is given by the following equation:

$$I_{DG} = 2.90 \times 10^{-5} \, SF \, f_{on} \tag{4-10}$$

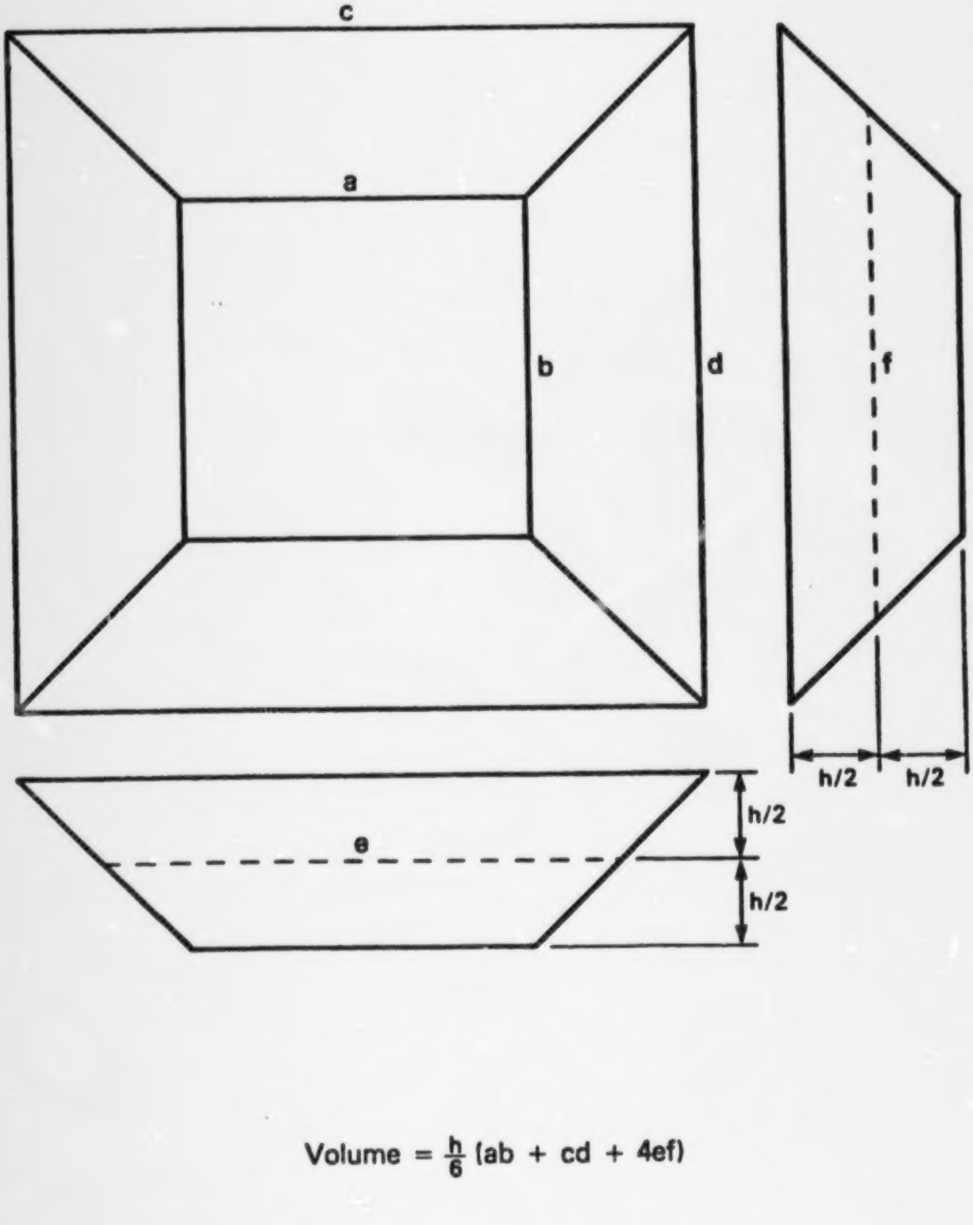
where f_{on} is given by equation 4-3 provided that TDEL exceeds the time it takes for the source waste stream to be accessed, f_{sc} (see Section 4.1.3). Otherwise, f_{on} is assumed to be zero. The factor 2.90×10^{-5} is just the factor 9.39×10^{-7} divided by the surface area of the assumed drill hole, i.e., 324 cm^2.

4.2.2 Intruder-Construction Scenario

This scenario assumes that at some time after the end of operations at the disposal facility, institutional controls breakdown temporarily and an intruder inadvertently constructs a house on the disposal facility. In so doing, the intruder is assumed to contact the disposed wastes while performing typical excavation work such as installing utilities, putting in basements, and so forth. These typical activities should not be expected to involve significant depths - e.g., in most cases no more than about 3 m (about 10 ft). There is, however, a much less likely chance that some excavation could proceed at a lower depth. This could occur, for example, through construction of a sub-basement for a high rise building.

To implement this scenario, the inadvertent intruder is assumed to dig a 3 meter deep cellar for the house. The surface area of the house is assumed to be 20 m by 10 m (200 m^2), which is a typical surface area for a reasonably large ranch-style house. The excavation for the cellar is assumed to be 20 m by 10 m (200 m^2) at the bottom and 26 m by 16 m at the top (giving a 1:1 slope for the side of the hole). This results in a total volume (see Figure 4.3) of excavated material equal to 906 m^3, of which a portion will be composed of the cover that was emplaced over the waste during disposal operations (675 m^3), a portion will be composed of waste, and the remainder will be composed of soil or other backfill mixed with the waste during disposal operations.

It is conservatively assumed during this scenario that the inadvertant intruder spends most of his time inside the excavation. Consequently, the


c
a
b
d
f
h/2
h/2
e
h/2
h/2

$$\text{Volume} = \frac{h}{6}\,(ab + cd + 4ef)$$

Figure 4.3. Foundation Volume Calculations

intruder exposures are calculated assuming a source term consisting of the waste and soil mixture within the disposal cell rather than the excavated material (see Section 4.2.4). The fraction of the waste/soil mixture that actually consists of waste is therefore given by the product of two factors: the surface utilization efficiency (SEF), and the waste emplacement efficiency (EMP).

The surface utilization efficiency (SEF) is a property of the disposal technology (see ID index - Section 2.2.3), and must be included when determining the fraction of the excavated soil/waste mixture that actually consists of waste. SEF is defined as the surface area of the disposal cells divided by the surface area of the disposal cells plus any space between the disposal cells. SEF is therefore a fraction less than or equal to unity.

The waste emplacement efficiency (EMP) is the fraction of the volume within a given disposal cell (excluding the cover) which consists of waste.

4.2.2.1 Routine Wastes

The equation describing human exposure for the intruder-construction scenario is as follows (Ref. 3):

$$H = H_{air} + H_{DG} \qquad (4\text{-}11)$$

where H is the 50-year dose commitment in mrem, and H_{air} and H_{DG} denote the contributions from the air uptake pathway and direct gamma pathways, respectively. The first term calculates impacts from air pathways resulting from suspension of contaminated dust into the air. These impacts include inhalation of the contaminated dust and direct radiation exposure from immersion in the contaminated dust cloud. The second term calculates the impacts from direct gamma radiation emitted by the excavated wastes. These two terms are given by the following equations:

$$H_{air} = \sum_n I_{air} C_n \text{ PDCF-2} \qquad (4\text{-}12)$$

$$H_{DG} = \sum_n I_{DG} C_n \text{ PDCF-5} \qquad (4\text{-}13)$$

where PDCF-2 and PDCF-5 are the pathway dose conversion factors which are discussed and presented in Appendix D, C_n is the radionuclide concentration in the waste, I_{air} and I_{DG} are the air and direct gamma pathway intraction factors, respectively, and n denotes the radionuclide.

Air uptake pathway interaction factor is given by the following equation:

$$I_{air} = f_o f_d f_w f_s \qquad (4\text{-}14)$$

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The time delay factor (f_o) is radionuclide-specific and is given by the following equation (4-3).

$$f_o = \exp[-\lambda \, TDEL] \qquad (4\text{-}3)$$

where λ is the decay constant of the radionuclide and TDEL is the time period between the end of active disposal operations and the initiation of the scenario. The assumed time period is equivalent to the assumption that the intrusion scenario involves the waste last disposed at the site and conservatively neglects the possibility that the intrusion scenario may involve waste disposed earlier during disposal facility operation.

The site design and operation factor, f_d, denotes the dilution of the waste due to particular disposal practices regarding waste emplacement. For routine and activated metal wastes, it is given by the product of two properties of the disposal technology: the waste emplacement efficiency (EMP) and the surface utilization efficiency (SEF), i.e.,

$$f_d = EMP \; SEF \qquad (4\text{-}15)$$

The waste form and package factor (f_w) is given by the factor f_D, the dispersibility multiplier (see Section 4.1.3).

The site selection factor is the product of the exposure duration factor (the fraction of a year that construction takes place) and the soil-to-air transfer factor T_{sa} (which depends on the environmental characteristics of the region in which the disposal facility is located).

In this work, the exposure duration is assumed to be 500 working hours. This is equivalent to a construction period of 3 months, which is believed to be reasonably conservative for typical construction. It is believed to be very conservative for activities involving use of heavy construction equipment. This gives a value of 0.057 for the exposure duration factor.

The soil-to-air transfer factor given by the formula:

$$T_{sa} = [T_{sa}]_o \times (10/v) \times (s/30) \times (50/PE)^2 \qquad (4\text{-}16)$$

where $[T_{sa}]_o$ is equal to 2.53×10^{-10}, v is the average wind speed at the site in m/sec, s is the silt content of the site soils in percent, and PE is the precipitation-evaporation index of the site vicinity indicative of the antecedent moisture conditions (see Reference 2). Parameter values for v,s, and PE are given for each reference site in Appendix C. For the southeast site, for example, values of $v = 3.61$ m/sec, $s = 50$, and $PE = 91$ result in a T_{sa} value equal to 3.53×10^{-10}. Multiplying T_{sa} by the density of the soil, which is assumed to be 1.6 gm/cm^3, gives the average airborne dust loading for the construction activity, while multiplying T_{sa} by 0.057 gives the site selection factor. The resulting values for each of the four reference sites are given in Table 4-1.

TABLE 4-1 . Reference Dust Loadings and Site Selection Factors

Site Environment	Dust Loading(mg/m^3)	Site Selection Factor - f_s
Northeast	0.258	9.18×10^{-12}
Southeast	0.565	2.01×10^{-11}
Midwest	0.705	2.51×10^{-11}
Southwest	7.410	2.64×10^{-10}

In summary, the interaction factor for routine wastes is given by

$$I_{air} = f_o \, f_d \, f_D \, f_s \tag{4-17}$$

where the symbols are as explained before, and where f_d is given by equation 4-15.

Direct radiation pathway interaction factor is given by the analogue of equation 4-14. The factors f_o and f_d are identical to those for the air pathway. Only the self-shielding of the particular waste form could conceivably affect the waste form and package factor f_w. For routine wastes, f_w is set equal to unity and the site selection factor, f_s, is assumed to be equal to the exposure duration factor 0.057. Thus the interaction factor is given by

$$I_{DG} = f_o \, f_d \, 0.057 \tag{4-18}$$

where the symbols are as explained before, and where f_d is given by equation 4-15.

4.2.2.2 Activated Metal Wastes

For these wastes, equations 4-11 through 4-13 for human exposures are applicable. The only difference for activated metals occurs in the waste form and package factor, f_w. For air uptake pathways, f_w is equal to the activated metal dispersibility multiplier f_{AD}. For the direct radiation pathway, it is equal to the self-shielding factor, f_{AG} (see Section 4.1.3). Thus the interaction factors are given by

$$I_{air} = f_o \, f_d \, f_{AD} \, f_s \tag{4-19}$$

$$I_{DG} = f_o \, f_d \, f_{AG} \, 0.057 \qquad (4\text{-}20)$$

where the symbols are as explained before.

4.2.2.3 Source Waste Streams

Again, only the waste form and package factor is affected for these waste streams. However, the variable C_n denotes the average total activity for each source. Several other modifications must also be considered. For these wastes, equation 4-11 is still applicable; however, the air and direct gamma contributions are given by the following:

$$H_{air} = \sum_n I_{air} \, C_n \, \text{PDCF-2} \qquad (4\text{-}21)$$

$$H_{DG} = \sum_n [I_{DG1} \, C_n \, \text{PDCF-5} + I_{DG2} \, C_n \, \text{PDCF-8}] \qquad (4\text{-}22)$$

where C_n is the average total activity per source (in Ci), PDCF-2, PDCF-5 and PDCF-8 are the pathway dose conversion factors (in units of mrem/yr per Ci/m^3, mrem/yr per Ci/m^3, and mrem/yr per Ci/m^2, respectively) which are discussed in Section 2.3, and I_{air}, I_{DG1}, and I_{DG2} (in units of m^{-3}, m^{-3}, and m^{-2}, respectively) are the interaction factors discussed below.

For the air uptake pathways, it is assumed that only the portion of the waste that is dug into is affected. Moreover, only those source waste streams within this volume that have delay times less than the time period between the closure of the site and the initiation of the scenario are considered. (Placement of the source within a special high integrity container would result in the package/source being dug up intact and would result in no dispersion of material.) Thus, the exposures are calculated using a factor f_w equal to the dispersibility multiplier f_{SD} times the volume density of the sealed sources, i.e., d_S, the number of sources per unit volume of waste in the waste class (see equation 2-4). The interaction factor is given by

$$I_{air} = f_o \, d_S \, f_{SD} \, f_s \qquad (4\text{-}23)$$

where f_o is assumed to be given by equation 4-3 if TDEL > f_{SC}; otherwise, f_o is equal to zero.

For the direct exposure pathways, it is assumed that there are two contributing components: source waste streams that are distributed within the undug volume (I_{DG1}) plus one source assumed to be lodged at the surface of the excavation (I_{DG2}). The assumed interaction factors are given below:

$$I_{DG1} = f_o \, d_S \, f_{SG} \, 0.057 \qquad (4\text{-}24)$$

$$I_{DG2} = f_o \, f_{SG} \, 0.057 \, (1/260) \qquad (4\text{-}25)$$

where f_o is always given by equation 4-3, f_{SG} is the self-shielding factor for the source (if any), d_S is the density of the source (in m^{-3}) calculated internally, and (1/260) expresses the distribution of the source over the exposed surface area of the waste (in m^{-2}).

The first equation models exposures as resulting from the distribution of a series of point sources throughout the volume of disposed waste. The second equation is based on the assumptions that (1) the source may be lodged at any spot on the floor or walls of the excavation (one location is not more likely than another), and (2) it is impossible to predict the likelihood of the intruder being in any particular location. The intruder may be at varying distances from the lodged source for varying lengths of time. Therefore, an equivalent exposure to the intruder may be calculated by assuming that the activity within the source is smeared over the total exposed surface area (260 m^2) of the waste/soil mixture within the excavation.

4.2.2.4 Scenario Application

Two final considerations relate to the applicability of this scenario: (1) depth of disposal, and (2) waste and disposal cell stability.

As noted above, typical construction excavations are normally thought to require relatively insignificant depths (e.g., 3 m or so), although there is some chance of excavations at greater depths (e.g., for high rise buildings). The greater the depth at which the waste is disposed, the less chance that the intruder-construction scenario will involve contact with the waste. The scenario would eventually be totally inapplicable. It seems reasonable to incorporate this principle into calculation of potential impacts, although it is not precisely clear how to do it. One approach would be to assume a strictly linear relationship between disposal depth and radiological impacts, although the uncertainties inherent in postulating future events make this approach seem overly prescriptive.

The approach adopted in this work is to assume some relatively simple limits in the application of the intrusion scenario based on requirements imposed on disposal depth. If, for a given waste stream, there are no minimum disposal depth requirements, then the full intrusion scenario is assumed. For disposal depths of at least 5 m, impacts are assumed to be reduced by a factor of 10. This could be accomplished, for example, by what has been described in Reference 2 as "layering," i.e., preferential placement of waste at lower levels of a disposal cell. Below 10 m, the waste is assumed to be essentially precluded from access from the intruder-construction scenario. For such waste the only applicable intrusion scenario is assumed to be the intruder-drilling scenario.

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The second consideration relates to the applicability of the scenario with regard to waste and disposal cell stability. The intruder-construction scenario is assumed to be only applicable to situations in which the waste is sufficiently decomposed to resemble ordinary soil. This could happen, for example, for cases in which waste is either disposed in a structurally unstable form, or is disposed mixed with structurally unstable waste. This point is discussed in greater detail in Section 4.2.3.

4.2.3 Intruder-Discovery Scenario

The intruder-discovery scenario is conceptualized as a modification of the intruder-construction scenario. It is based on the assumption that the operational practice of segregated disposal of stable waste streams from unstable waste streams results in reduced exposures to a potential inadvertant intruder contacting the stable waste streams -- at least for the first several hundred years following waste disposal. Segregated disposal of the stable waste streams greatly improves the stability of disposal cells containing the stable wastes, resulting in significantly less water infiltration and subsidence for these disposal cells, and less decomposition of the disposal cell contents. Exposures to a potential inadvertant intruder excavating into these disposal cells at the end of the active institutional control period would be limited to those received during discovery of the waste. It is not credible, for example, to postulate that an intruder would construct a house in, or attempt to grow vegetables in, a disposal cell composed of such wastes as 55-gallon drums filled with concrete. Neither is it credible to assume an extensive intruder-construction scenario for grouted waste or for waste disposed in reinforced concrete structures.

To implement this scenario, the same activities are assumed as for the intruder-construction scenario, except that the exposure duration time is reduced from 500 working hours to 6 hours. This accounts for the time spent by the intruder while excavating into and discovering the waste, and deciding what to do about it. Upon discovering the waste, it is assumed that the intruder consults records or other data, or puts out an open solicitation to several consulting firms, to try and determine what the material could be and who left it. During the course of these investigations, it is assumed that the intruder determines that the site had once been used for disposal of radioactive waste. This is believed to be reasonable since inadvertent intrusion for at least several centuries following waste disposal will most likely be the result of a bureaucratic error rather than a deliberate release of the site for unrestricted use.

It could be argued that the inadvertant intruder could spend more than 6 hours contacting the waste. The possibility of additional time, however, is balanced by the conservative use of the same areal extent of the excavation as that for the intruder-construction scenario. That is, the surface area for the construction activity is still assumed to be 26 m by 16 m = 416 m^2. In reality, the amount of waste contacted would probably only cover a few square meters prior to the realization by the intruder that he is contacting something unusual. The smaller exposed area would result in reduced inhalation and direct gamma impacts.

It should also be noted that the inadvertent intruder would contact the topmost layer of waste first, and would likely stop excavating before digging too far into the rest of the waste. Thus, stable waste placed at the bottom of the disposal cell (layered waste) is assumed in this scenario not to be contacted by the intruder. It should also be noted that the intruder-discovery scenario preempts the following intruder-agriculture scenario.

Finally, consideration needs to be given to the length of time that the intruder-discovery scenario would be applicable. That is, how long will segregated stable wastes deter or reduce impacts to the inadvertant intruder? Based on the analysis in the draft Part 61 EIS (Ref. 12), a time period of 500 years after site closure is used as a limit of effectiveness of stable waste segregation. Following this time period, the full intruder-construction (and the following intruder-agriculture) scenario is assumed for segregated stable waste. A similar assumption is applied for grouted waste or for waste disposed in reinforced concrete structures.

Layered waste, e.g., Class C or other waste disposed with 5-10 m of cover by lower activity waste or other material, is still assumed to be difficult to contact even after 500 years. A factor of 10 reduction in impacts is assumed for layered waste when implementing either the intruder-construction or intruder-agriculture scenario after 500 years.

4.2.4 Intruder-Agriculture Scenario

It is assumed in this scenario that an intruder lives in the building constructed as part of the intruder-construction scenario, and consumes food grown in soil contaminated by the decomposed waste excavated and distributed around the building during construction activities.

To implement the scenario, a portion of the soil and waste (including the earthen cover) excavated during the intruder-construction activity is assumed to be distributed around the completed house. After building the foundations of the house, about 306 m^3 of this material would be put back outside and around the cellar walls leaving a volume of about 600 m^3 of material which is distributed around the site property.

The precise areal extent to which this material is distributed is somewhat speculative. It is likely, however, that the material will remain localized; moving even a few cubic yards of soil more than 10 meters usually requires significant effort involving use of expensive heavy machinery. It is assumed in this report that this areal extent is likely to be somewhere between 1000 m^2 and 2000 m^2. That is, the waste/soil mixture is assumed to lie within a radius of 25 m from the house.

The intruder is then assumed to live in the building constructed in this distributed waste/soil mixture, work at a regular job during the day, and spend some of his extra time working in a garden growing food for his own consumption. His time during a year is assumed to be allocated between various activities as given in Table 4-2.

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TABLE 4-2 . Intruder Time Allocation

Activity	Hours/Year
At Home	4380
At Work	2000
Traveling To/From Work	250
Vacation	330
Gardening	100
Outdoors	1700

In the intruder-agriculture scenario, the inadvertent intruder could be exposed principally by five pathways: (1) inhalation of contaminated dust suspended due to tilling and other activities as well as natural suspension, (2) direct radiation exposure from standing in the contaminated cloud, (3) consumption of food (e.g., leafy vegetables) dusted by fallout from the contaminated cloud, (4) consumption of food grown in the contaminated soil, as well as products from animals fed contaminated forage, and (5) direct radiation exposure from the disposed waste volume.

For calculational convenience, the first three subpathways have been grouped together into an "air uptake" pathway. The potential exposures from these pathways are therefore calculated in three groups: air uptake, food (soil) uptake, and direct radiation (volume) exposures. These are then added to arrive at the total potential exposures from this scenario.

The potential exposures from the intruder-agriculture scenario are calculated using the following equation:

$$H = H_{air} + H_{food} + H_{DG} \tag{4-26}$$

where H is the annual dose rate in mrem per year during the 50th year of exposure, and H_{air}, H_{food}, and H_{DG} are the individual contributions from the air, food, and direct gamma subpathways. These are discussed below.

4.2.4.1 Routine Wastes

For these wastes, the contributions of the subpathways are calculated as follows:

$$H_{air} = \sum_n I_{air} C_n \text{ PDCF-3} \tag{4-27}$$

$$H_{food} = \sum_n I_{food} C_n \text{ PDCF-4} \tag{4-28}$$

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$$H_{DG} = \sum_n I_{DG} C_n \text{ PDCF-5} \qquad (4\text{-}29)$$

where the radionuclide is denoted by n; I_{air}, I_{food}, I_{DG}, are the interaction factors given by equation 4-2 for the air, food and direct gamma pathways, respectively; PDCF-3, PDCF-4, and PDCF-5 are the radionuclide specific pathway dose conversion factors presented in Appendix D; and C_n is the radionuclide concentration in the waste. The values of the transfer factors are presented below.

The time delay factor (f_o) for this scenario is identical with that for the intruder-construction scenario, and is always given by equation (4-3). The site design and operation factor (f_d) is determined in a similar manner as that for the intruder-construction scenario and is the same for all three of the above pathways. In addition, however, f_d includes the dilution resulting from mixing the excavated soil/waste with the excavated cover soil. Thus, f_d is equal to the product of three factors: (1) the surface disposal efficiency (SEF), (2) the waste emplacement efficiency (EMP), both of which were discussed previously, and (3) the cover mixing efficiency (RMIX) which is the fraction of the total volume of excavated material that consists of the soil/disposed waste mixture. The cover mixing efficiency can be calculated to be 0.25 using the methodology presented in Reference 3. Thus, f_d is given by

$$f_d = \text{EMP SEF } 0.25 \qquad (4\text{-}30)$$

For routine wastes, the waste form and package factors for the air uptake and direct radiation exposure pathways are identical with those for the air uptake and direct radiation exposure pathways of the intruder-construction scenario. However, for the food (soil) uptake pathway, other considerations are applicable. The following formula is utilized to calculate (f_w) for the food (soil) uptake pathway:

$$f_w = M_o \qquad (4\text{-}31)$$

where, M_o is the radionuclide-specific leach fractions of unsolidified waste forms (see Section 4.3). It appears to be reasonable to assume that only the fraction of radionuclides transferred from the waste to the interstitial water will be accessible to the roots. In addition, it would appear to be reasonable to include a contact time fraction in the above equation, i.e., the fraction of time in one year that the waste is in contact with water. The contact time fraction is conservatively assumed to equal unity, however, given a number of uncertainties including the extent of time that the garden is watered.

The site selection factor for the air uptake pathway is similar to that for the intruder-construction air uptake pathway. However, the soil-to-air transfer factor must be averaged to account for natural resuspension of the soils for part of a year. This estimate is calculated by assuming that (1) the airborne dust loading for the intruder-construction scenario (see Section 6.2.1) is applicable during gardening (100 hours); (2) during the time spent outdoors (1700 hours), a typical natural outdoor ambient air particulate concentration of 0.1 mg/m^3 is assumed (Ref. 4); and (3) during the time spent indoors (4348 hours), a typical ambient indoor concentration of 0.05 mg/m^3 is assumed (Ref. 4). Time-averaging these values results in the site selection factor values given in Table 4-3.

TABLE 4-3 . Reference Site Selection Factors for the Intruder-Agriculture Scenario

Site Environment	Site Selection Factor
Northeast	$2.96\text{x}10^{-11}$
Southeast	$3.18\text{x}10^{-11}$
Midwest	$3.28\text{x}10^{-11}$
Southwest	$8.06\text{x}10^{-11}$

For the food (soil) uptake pathway, f_s is assumed to be the fraction of food consumed by the individual that is grown on site, 0.5.

For the direct radiation exposure pathway, f_s is equal to the exposure duration fraction multiplied by a correction factor to account for the limited areal extent of the source of direct radiation. Moreover, the fraction of the time the intruder spends in relation to the source must be considered. During a year, the intruder is assumed to spend 1800 hours outdoors exposed to unattenuated radiation (100 hours tilling and 1700 hours around the house). During the 4380 hours he spends indoors, he is exposed to attenuated radiation. The correction factor due to the limited areal extent of the radiation source may be estimated using Figure 4.4.

This figure shows that intruder may be assumed to be exposed to a full disk source while outside the house, and an annular source while inside the house. While he is inside the house, the center of the disk represents the shielding provided by the foundation slab. The contribution to the direct radiation exposure from this center portion may be neglected in comparison with the exposure from the outside of the house. If the foundation slab is a one-foot thick concrete layer, the radiation would be attenuated to about 0.03 of its unshielded value for Cs-137 gamma rays (Ref. 11). The correction factor for the areal extent of the annular source may be represented by the following equation:

$$CF = [E_1(ur_1) - E_1(ur_2)] / E_1(ur_0) \tag{4-32}$$

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where CF is the dimensionless correction factor, $E_1(x)$ is the first order exponential integral, u is the linear attenuation coefficient of air in m^{-1} (it is taken to be 0.0097 m^{-1} in this report) (Ref. 11), and the r's are the distances from the exposure point indicated in Figure 4.4 in m.

In order to evaluate the correction factor, these radial distances must be assumed. Values of the exponential integral for some representative distances are given in Table 4-4.

TABLE 4-4 . Exponential Integral Values

Distance	ur	$E_1(ur)$
1 m	0.0097	4.068
8 m	0.0776	2.055
20 m	0.1940	1.248
25 m	0.2425	1.068

For an annular source (the time spent indoors), it is reasonable to assume 1 m and 8 m, respectively, for r_0 and r_1; 1 m represents the effective height of the exposed person, and 8 m represents the approximate radius of a 200 m^2 house floor. The value assigned to r_2, however, depends on the areal extent to which the waste/soil mixture (600 m^3) has been spread. This mixture will likely be spread unevenly within about a half acre around the house excavation, and the areal extent is likely to be between 1000 m^2 and 2000 m^2. A radius of 20 m represents an area of about 1050 m^2 over which the waste is spread, while a radius of 25 m represents an area of about 1750 m^2. A radius of 25 m is utilized in this work. For a full disc source (the time spent outdoors), r_2 is again assumed to be 25 m. However, the radius r_0 in equation 4-32 is replaced by r_1.

These assumptions yield a correction factor for the time spent outdoors of about 0.74, and a correction factor for the time spent indoors of about 0.24. Utilizing values of 1800 hours outdoors and 4248 hours indoors yields a site selection factor of about 0.27, which is the value utilized in this report for direct radiation exposure.

In summary, the following interaction factors are used for routine wastes:

$$I_{air} = f_o \, f_d \, f_D \, f_s \qquad (4\text{-}33)$$

$$I_{food} = f_o \, f_d \, M_o \, 0.5 \qquad (4\text{-}34)$$

$$I_{DG} = f_o \, f_d \, 0.27 \qquad (4\text{-}35)$$

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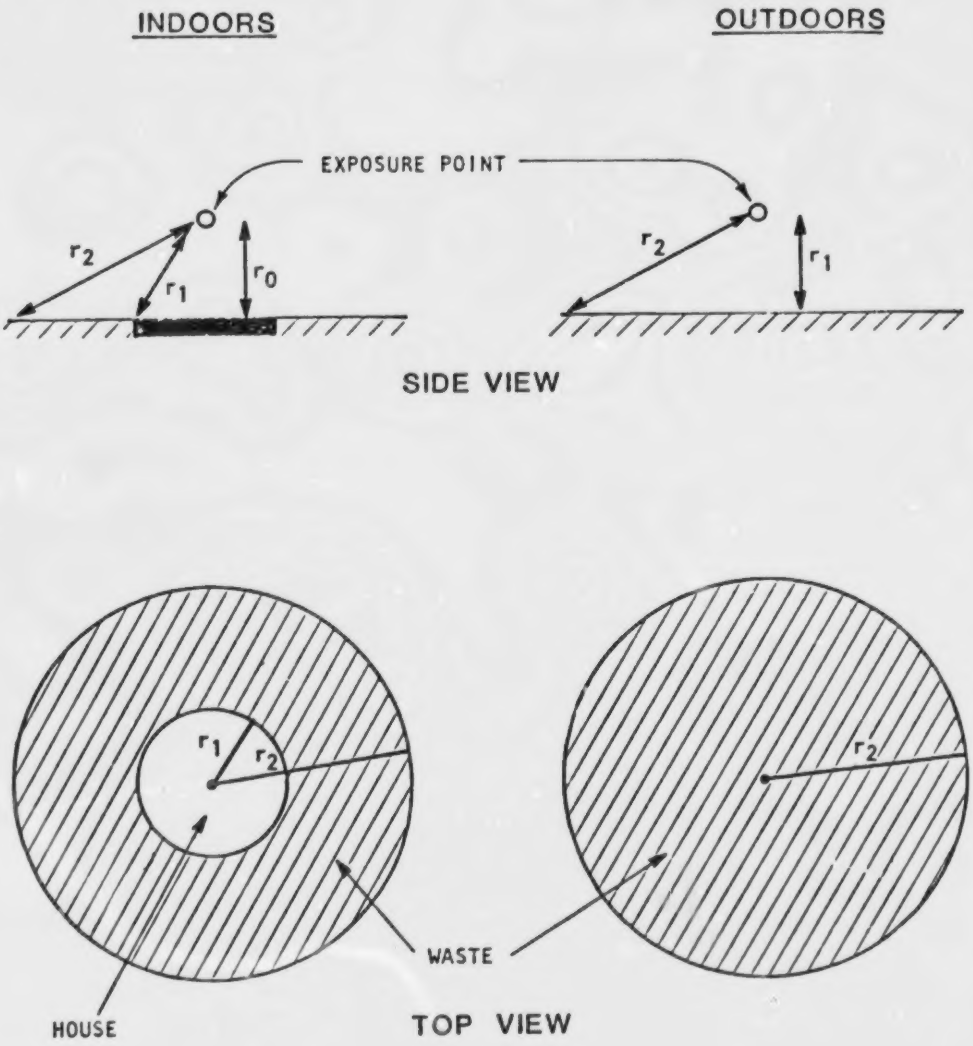
INDOORS
OUTDOORS
EXPOSURE POINT
r_2
r_1
r_0
r_2
r_1
SIDE VIEW
r_1
r_2
r_2
WASTE
HOUSE
TOP VIEW

DIRECT RADIATION EXPOSURE GEOMETRY

FIGURE 4.4

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where f_d is given by equation (4-30), and where all the other symbols are as explained above.

4.2.4.2 Activated Metal Wastes

For these wastes, the following equations give the contributions from the subpathways:

$$H_{air} = \sum_n I_{air} C_n \text{ PDCF-3} \qquad (4\text{-}36)$$

$$H_{food} = \sum_n I_{food} C_n \text{ PDCF-4} \qquad (4\text{-}37)$$

$$H_{DG} = \sum_n [I_{DG1} C_n \text{ PDCF-5} + I_{DG2} C_n \text{ PDCF-8}] \qquad (4\text{-}38)$$

where n stands for the radionuclide; C_n is the radionuclide concentration in Ci/m^3; PDCF-3, PDCF-4, PDCF-5, and PDCF-8 are the pathway dose conversion factors discussed in Appendix D; I_{air}, I_{food}, I_{DG1}, and I_{DG2} are the interaction factors discussed below.

The major consideration for these wastes is the delay time afforded by corrosion. The activity in the corroded portion of the waste streams is distributed through the entire 600 m^3 of waste and soil spread around the house vicinity. This corroded material contributes to the scenario impacts from all three subpathways. For the air and food subpathways, interaction factors are given by the following:

$$I_{air} = f_o f_d f_{AD} f_s \qquad (4\text{-}39)$$

$$I_{food} = f_o f_d f_{AW} M_o 0.5 \qquad (4\text{-}40)$$

where f_d is given by equation (4-30), and where f_{AD} and f_{AW} are discussed in Section 4.1.3. For the direct gamma scenario, there are two distinct contributions. The portion of the waste that is corroded is distributed over the volume of waste affected; and the portion that remains intact contributes in a manner similar to source waste streams. The interaction factors are given by:

$$I_{DG1} = f_o f_d f_{AD} f_{AG} 0.27 \qquad (4\text{-}41)$$

$$I_{DG2} = f_o f_d (1-f_{AD}) f_{AG} 0.132 \; 0.27 \qquad (4\text{-}42)$$

where f_o is the time delay factor, f_{AD} and f_{AG} are the activated metal dispersibility and self-shielding factors, respectively (see Section 4.1.3), 0.27 is the site selection factor discussed above, and 0.132 is the ratio of the volume of waste/soil excavated during the intruder-construction scenario (231 m^3) to the area over which the exposure occurs (1750 m^2).

4.2.4.3 Source Waste Streams

These wastes again require special consideration. The three subpathway equations are given by

$$H_{air} = \sum_n I_{air} C_n \text{ PDCF-3} \tag{4-43}$$

$$H_{food} = \sum_n I_{food} C_n \text{ PDCF-4} \tag{4-44}$$

$$H_{DG} = \sum_n [I_{DG1} C_n \text{ PDCF-5} + I_{DG2} C_n \text{ PDCF-8}] \tag{4-45}$$

where n stands for the radionuclide; C_n is the radionuclide activity in Ci; PDCF-3, PDCF-4, PDCF-5, and PDCF-8 are the pathway dose conversion factors discussed in Appendix D; I_{air}, I_{food}, I_{DG1}, and I_{DG2} (in units of m^{-3}, m^{-3}, m^{-3}, and m^{-2}, respectively) are the interaction factors discussed below.

Only the portion of the sources that are excavated are assumed to contribute to the subpathways. The sources may be dug up in either of two ways: before the source interaction delay time f_{SC} when they cannot be accessed by transfer agents, and after this delay time when they can be assumed to be accessible.

In the former case, the sources do not contribute to either the air or the food subpathways and only the direct gamma exposures result; however, it is conservatively assumed that all the sources are at the surface of the excavated material spread around the house vicinity. The interaction factors are given by the following:

$$I_{air} = I_{food} = I_{DG1} = 0 \tag{4-46}$$

$$I_{DG2} = f_o \, (231 \, d_S) \, (1/1750) \, f_{SG} \, 0.27 \tag{4-47}$$

where f_o is as discussed above, 231 is the volume of the waste/soil mixture (not including the cover soil) excavated during the intruder-construction scenario (in m^3), d_S is the density of the source (in m^{-3}) within the disposal cell (the product 231 x d_S is equal to the total number of sources accessed), f_{SG} is the self-shielding factor for the source (if any), and (1/1750) expresses the distribution of the source over the exposed surface area of the waste (in m^{-2}).

In the latter case, those source waste streams that have delay times less than the time period between the closure of the site and the initiation of the scenario are considered to be distributed within the entire 906 m^3 of waste and cover soil excavated during the intruder construction scenario. Thus, the site design and operation factor, f_d, becomes the total number of sources accessed, ($231 \times d_S$), divided by 906 m^3:

$$f_d = 231 \times d_S / 906 \tag{4-48}$$

For the source waste streams, the following equations give the contributions from the three subpathways:

$$I_{air} = f_o \, f_d \, f_{SD} \, f_s \tag{4-49}$$

$$I_{food} = f_o \, f_d \, f_{SW} \, M_o \, 0.5 \tag{4-50}$$

$$I_{DG1} = f_o \, f_d \, f_{SG} \, 0.27 \tag{4-51}$$

and

$$I_{DG2} = 0 \tag{4-52}$$

where f_{SD}, f_{SW}, and f_{SG} are discussed in Section 4.1.3, and the other terms are as explained before.

4.2.4.4 Drilling Wastes

A final source of radiation exposure will be from the radioactive material excavated as part of the intruder-drilling scenario. As discussed in Section 4.2.1, radioactive material is assumed to be excavated as part of a hypothetical event in which a water well is drilled for a house. The excavated material, along with drilling mud and other drill cuttings, are contained within a mud pit dug by the drilling crew. After the well is drilled, the mud pit is filled in with soil, the well installation process is completed, and housing construction is commenced.

It is somewhat difficult to postulate a reasonable scenario by which persons living at the house thus constructed will be exposed to significant radiation. It could be argued that the radioactive material in the mud pit could be excavated and distributed around the vicinity of the house as part of landscaping activities. This is believed to be rather unlikely, however. The well would be dug well away from the vicinity of the house for a number of common sense reasons. Landscaping activities would typically involve use of heavy machinery such as bulldozers, and one would obviously want to keep away from the well in order not to damage it.

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Another potential pathway could involve root uptake of crops grown in the garden established as part of the intruder-agriculture scenario. Again, this is believed to be unlikely to constitute a significant exposure pathway. The immediate vicinity of a well does not seem to be a particularly good place to put a garden. In any event, the surface area of the mud pit is only about 72 ft^2 (6.69 m^2), which means that the number of food plants that could be affected by the buried waste would be relatively small.

The most important exposure pathway would appear to involve direct gamma radiation as attenuated through a soil cover. In this case, individual exposures may be calculated as follows:

$$H = \sum_n I_{DG} C_n \text{ PDCF-5} \qquad (4\text{-}6)$$

where H is the 50-year dose commitment in mrem/yr, the radionuclide is denoted by n, I_{DG} is the interaction factor for the direct gamma pathway, C_n is the concentration in the waste accessed by the drill, and PDCF-5 is the radionuclide-specific pathway dose conversion factor.

Routine Wastes

The interaction factor I_{DG} for wastes other than activated metals and sources is given by the following:

$$I_{DG} = 1.636\times10^{-2} \text{ EDF CF SF DTK EMP } f_{on} \qquad (4\text{-}53)$$

where the factor 1.636×10^{-2} (in m^{-1}) is the product of the area of the drill hole (324 cm^2) divided by the dilution volume (1.98 m^3).

EDF is the exposure duration factor, or the fraction of a year that the exposed individual spends in close proximity to the radiation source. In the case of the well, there is no reason to suppose that the exposed individual will spend any more time near the well than anywhere else on the property. Water wells are almost all automated (and will be more so in the future), and there would be no special need to make regular trips out to it. Due to the effects of radiation attenuation through soil, the exposed individual would receive little or no exposure until he or she was in very close proximity to the source -- i.e., practically standing on top of it.

In this report, the exposed individual is conservatively assumed to spend 1700 hours outside and within a 25 m radius of the house. The outside area within this radius is 1750 m^2, and if it is assumed that the probability of being in one particular location on this area is the same as being in any other location, then the amount of time spent near the radiation source can be estimated by using a ratio of the outside and mud pit areas. That is, the mud pit cross-sectional area is 6.69 m^2, and so the number of hours spent in close proximity to the mud pit is 1700 x 6.69/1750 = 6.5 hours. EDF is then calculated to be 6.5/8760 = 7.417E-4.

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CF, the areal exposure correction factor, is calculated using the procedures set forth in Chapter 3 of Reference 3. In this reference, CF values are calculated and listed for varying distances from an infinite slab radiation source having a variable radius. The radius of a circle having the cross-sectional area of the mud pit is 1.46 m. Assuming a source radius of 1.46 m and an exposure distance of 1 m, CF is calculated to be 0.0917.

The factor SF is the shielding factor for a 2-ft thick layer of soil. It is calculated using equation 4-7 as discussed in Section 4.2.1.1. The only difference is that radionuclide-specific linear attenuation coefficients corresponding to SiO_2 are used rather than those for water (Ref. 11).

The variables DTK (in meters) and EMP (dimensionless) are the disposal technology specific variables discussed in Section 2.2.3. The factor f_{on} is given by equation 4-3. Combining terms, I_{DG} reduces to:

$$I_{DG} = 1.11x10^{-6} \text{ SF DTK EMP } f_{on} \tag{4-54}$$

Activated Metal Wastes

The interaction factor I_{DG} is also given by the above equation except that the characteristics of the waste form are considered when calculating f_{on}. f_{on} is calculated by equation 4-3 provided that TDEL exceeds the time it takes for the metal to corrode completely, i.e., f_{AC} as discussed in Section 4.1.3. If f_{AC} exceeds TDEL, however, f_{on} is assumed to be zero. This corresponds to the assumption that drilling would not occur through intact metal.

Source Waste Streams

For source waste streams, the interaction factor is somewhat modified. As in Section 4.2.1.3, the variable C_n for source waste streams stands for the total activity of the source accessed by the drill, with the result that the interaction factor I_{DG} has dimensions of m^{-3}. I_{DG} is given by the following equation:

$$I_{DG} = 3.43x10^{-5} \text{ SF } f_{on} \tag{4-55}$$

where the factor $3.43x10^{-5}$ is just the factor $1.11x10^{-6}$ divided by 0.0324, which is the cross-sectional area of the drill hole. SF is the shielding factor as given above for routine waste streams. The factor f_{on} is given by equation 4-3 provided that TDEL exceeds the time it takes for the source waste stream to be accessed, f_{SC} (see Section 4.1.3). Otherwise, f_{on} is assumed to be zero.

4.2.4.5 Scenario Application

The application of the intruder-agriculture scenario with respect to disposal depth and to waste and disposal cell stability is assumed to be similar to that for the intruder-construction scenario. That is, it is not considered for waste disposed in a stable manner provided that the time period that the scenario is considered is less than 500 years following the end of the surveillance period. This can include waste stabilized through waste form or packaging, waste disposed using a grout backfill, or waste disposed in reinforced concrete bunkers. The intruder-agriculture scenario is furthermore fully considered only for waste disposed at a location less than 5 m below the earth's surface. For waste disposed between 5 and 10 m below the earth's surface, the scenario is assumed to be only 10% effective. For waste disposed at depths greater than 10 m, the intruder-agriculture scenario is considered only for the small quantities of waste brought to the surface during the intruder-drilling scenario.

4.2.5 Radionuclide Chains

This section considers the effects of radionuclide chains and daughter ingrowth on the intruder scenarios. Two major effects are considered in this section: (1) radionuclide chain daughters that decay into being during TDEL, i.e., the time between the end of disposal operations and the initiation of the intruder scenarios, and (2) the effects of radon-222 ingrowth during TDEL. The first effect would appear to include the second effect; however, since radon-222 is a gas, additional equations must be developed. Thus, it is considered as a separate effect.

These effects are applied to impact calculations involving exposures to individual inadvertent intruders, off-site exposures to populations due to the intrusion, and hypothetical impacts from erosion.

4.2.5.1 Effects of Radionuclide Chain Daughters

The first effect is relatively straightforward to incorporate. Taking TDEL as the decay time, Bateman decay equations have been utilized to calculate the concentrations of the daughter products. There are four natural radioactive decay chains, with natural precursors being U-238 (4n+2 chain), U-235 (4n+3 chain), Th-232 (4n chain), and Np-237 (4n+1 chain). In addition, there are several chains that have artificial radionuclides as precursors that merge with one of these four natural chains after several decay steps, e.g., Am-241 merges with the Np-237 chain, Pu-239 merges with the U-235 chain, etc. All these chains have been considered and significant relationships have been incorporated into the codes.

However, there is one minor complication in this scheme, and it involves the solubility class of the daughter products. In general, the solubility class of a radionuclide is a characteristic of its chemical form. Radioactive decay is invariably thought of as being a process that destroys chemical bonding. Experimental information on the chemical properties properties of the daughter products of chains does not appear to exist. Given these restrictions, assumptions have to be made on the solubilities of daughter products. In general, the most conservative chemical form is assumed for the daughter products. Assumed chain relationships are presented in Table 4-5.

TABLE 4-5 . Decay Chains Considered

4n			4n+1	4n+2		4n+3
Cm-244	Cf-252	Pu-236	Pu-241	Pu-242	Cm-242	Cm-243
Pu-240	Cm-248	U-232	Am-241	U-238	Pu-238	Am-243
U-236	Pu-244	Th-228	Np-237	U-234	U-234	Pu-239
Th-232	Pu-240		U-233	Th-230	Th-230	U-235
Ra-228	U-236		Th-229	Ra-226	Ra-226	Pa-231
Th-228	Th-232			Pb-210	Pb-210	Ac-227
	Ra-228					
	Th-228					

Each one of the chains presented in Table 4-5 are implicitly included in the codes. When the intruder impacts due to one of the radionuclides in Table 4-5 are being considered, the code searches for the chain that the radionuclide is a member of, calculates the daughter concentrations using Bateman decay equations, and includes the impacts due to the daughters as part of the impacts resulting from the precursor.

For isotopes belonging to decay chains, the revised methodology determines for a given decay time the fractional amounts of respective daughter nuclides which would be ingrown during the decay time periods. These fractions are then multiplied by the respective pathway dose conversion factors for the daughter nuclides, the products summed, and the total added to the pathway dose conversion factor for the parent nuclide.

That is, for the nuclides that do not belong to heavy metal decay chains, an entity f_o x PDCF is used in the calculations, where f_o is the time delay factor (see equation 4-3). For nuclides belonging to decay chains, this entity is replaced by the following:

$$f_o \ \mathrm{PDCF}_p + (f_{d1} \ \mathrm{PCDF}_{d1}) + (f_{d2} \ \mathrm{PDCF}_{d2}) + (f_{d3} \ \mathrm{PDCF}_{d3}) \ldots, \tag{4-56}$$

where

$$f_o = \exp(-\lambda T) \tag{4-57}$$

with

λ = decay constant of parent,
T = decay time period,
PDCF_p = PDCF of parent,

f_d = fractional quantity of a given daughter generated from a unit quantity of the parent at a given time, and
$PDCF_d$ = PDCF for a given daughter.

4.2.5.2 Effects of Radon Gas Generation

Some waste streams may contain small to moderate quantities of precursors of radon-222, in which case ingrowth of radon gas within buildings must be considered as part of the intruder-agriculture scenario. Equations to calculate impacts may be obtained from References 13 and 14. Assuming steady state conditions, the approach is to first calculate the radon flux from soil from a given concentration of radium-226 within the soil. Next, assuming that the structure built by the inadvertent intruder has concrete floors, the diffusion of radon gas through the concrete floor and into the structure is calculated. Then, assuming that at equilibrium the rate of radon gas diffusing into the building is equaled by the rate of radon gas lost by decay and other means, the average concentration of radon-222 within the building is calculated. Finally, this average concentration is multiplied by a factor which represents an individual's average yearly occupancy of the structure, and by an appropriate pathway dose conversion factor to determine individual impacts in mrem/yr.

The radon emanation rate from soil is given as follows (Ref. 13):

$$J_o = \sum_n C_n E_m \sqrt{\lambda D_s} \tag{4-58}$$

where

J_o = radon surface flux from soil (Ci/m^2-sec);
C_n = radium-226 concentration in soil (Ci/m^3);
E_m = emanation coefficient (assumed to be 0.2 - Ref. 13);
λ = radon-222 decay constant (2.1×10^{-6} sec^{-1}); and
D_s = effective diffusion coefficient through soil (assumed to be 2.2×10^{-6} m^2/sec - Ref. 13).

In addition, any radium-226 generated through the decay of thorium-230 is added to any radium-226 initially present in the waste. Higher order precursors are not considered because there is not sufficient time for their decay. For routine and activated metal waste streams, the variable C_n is rather straightforward. However, for source waste streams it is replaced by the term (C_n d_S) where C_n is the total activity per source and d_S is the total number of sources per unit volume.

The radon flux through a concrete floor with thickness t, J (in units of Ci/m^2-sec), and the radon-222 gas concentration within the building, C (in units of Ci/m^3), are given by (Ref. 14):

$$J = J_o \exp[-t \sqrt{\lambda / D_e}] \tag{4-59}$$

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$$C = J\ A\ /(V\ \lambda_e) \qquad (4\text{-}60)$$

where

J_o = radon flux calculated using equation 4-58;
λ^o = radon-222 decay constant;
D_e = effective radon diffusion coefficient for concrete (assumed to be $6\mathrm{x}10^{-9}$ m^2/sec - Ref. 13);
t = thickness of concrete (assumed to be 0.305 m - 1 ft);
A = surface area of the concrete floor (m^2);
V = volume into which the radon flux enters (m^3); and
λ_e = effective loss constant (decay plus removal) (assumed to be 1 hr^{-1} - Ref. 14).

Assuming as before a reasonably large ranch-style house, A is assumed to be 200 m^2. Assuming 8 foot high walls in the building, V is calculated to be about 500 m^3.

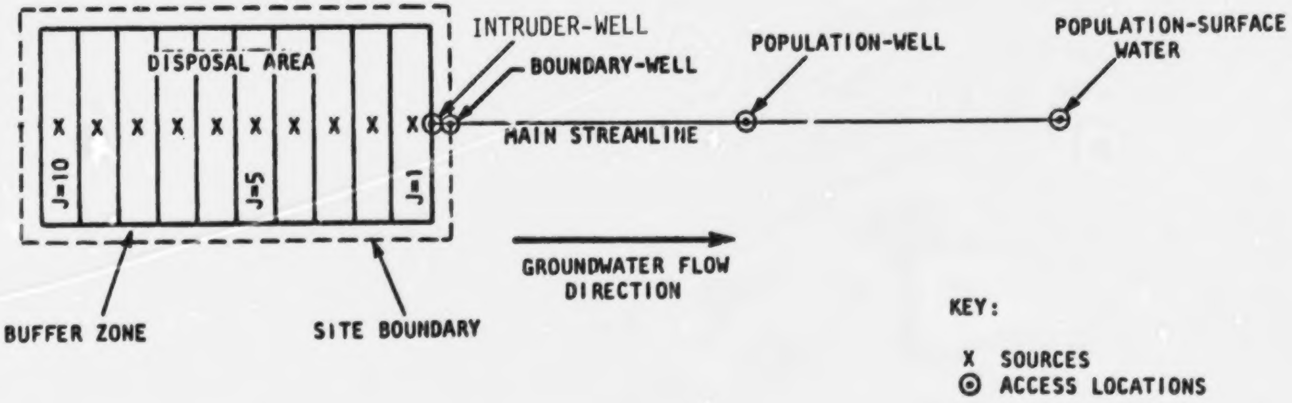
After calculating the average radon gas concentration (C) within the building, annual impacts are determined as:

$$H = \sum_i C\ EDF\ PDCF\text{-}3 \qquad (4\text{-}61)$$

where H is in mrem/yr and C is as discussed above. The exposure duration factor, EDF, is obtained from the assumed activity distribution presented at the beginning of this section for the individual intruder (i.e., the intruder spends half his time indoors, implying that EDF=4380/8760=0.5). PDCF-3 is the appropriate pathway dose conversion factor for radon-222. Impacts are summed over all waste streams containing radium-226 at the time that the intruder scenario is initiated, and added to the other impacts calculated for the intruder-agriculture scenario.

4.3 Groundwater Scenarios

These scenarios calculate the potential impacts resulting from groundwater migration of radionuclides from the disposed wastes to four biota access locations downstream in the direction of the groundwater flow: (1) a well located at the boundary of the disposal area, (2) a well located just outside the buffer zone of the disposal facility, (3) a well located between the disposal facility and the surface hydrologic boundary, and (4) a stream located at the surface hydrologic boundary. Different pathway dose conversion factors are used depending on whether the access location is a well or a stream (see Appendix D). An idealized map showing the geometric relationships between the disposal facility and the biota access locations is shown as Figure 4.5.


INTRUDER-WELL
BOUNDARY-WELL
POPULATION-WELL
POPULATION-SURFACE WATER
DISPOSAL AREA
X X X X X X X X X X
J=10
J=5
J=1
MAIN STREAMLINE
GROUNDWATER FLOW DIRECTION
BUFFER ZONE
SITE BOUNDARY
KEY:
X SOURCES
⊙ ACCESS LOCATIONS

GEOMETRIC RELATIONSHIPS
OF DISPOSAL AREA AND DISCHARGE LOCATIONS

FIGURE 4.5

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As shown in this figure, the main streamline passing underneath the disposal facility has been straightened out (the longitudinal coordinates are measured along this streamline). The overall area of the disposal facility, which influences the migration calculations, varies depending upon the particular facility considered. It may also be observed that the disposal area in Figure 4.5 has been divided into 10 sectors. In the calculation, the total radionuclide quantity disposed in the disposal site is divided by 10, and the resultant tenth is assumed to be released as a point source into each sector. The total impacts from migration from each point source are summed. The following equation is used to calculate human exposures which may result from the well access groundwater scenarios:

$$H = \sum_i \sum_n I_i \, C_{in} \, \mathrm{PDCF\text{-}6} \qquad (4\text{-}62)$$

where H is the annual dose rate in mrem per year during the 50th year of exposure, I_i is the interaction factor detailed below, PDCF-6 is the radionuclide-specific pathway dose conversion factor discussed and presented in Appendix D, C_{in} is the concentration of the radionuclide in the ith waste stream considered, i denotes the waste streams, and n denotes the radionuclides. For a surface water access location the dose conversion factor PDCF-7 is substituted instead of PDCF-6.

The interaction factor I_i is given by equation 4-5 and is composed of the four transfer factors:

$$I_i = f_o \, f_{di} \, f_{wi} \, f_{si} \qquad (4\text{-}63)$$

The time delay factor f_o and the site design and operation factor f_{di} are assumed to be unity. This merely means that the groundwater migration scenario is assumed to be initiated at the close of the the operational period and that the influence on the calculations resulting from design of the disposal facility is considered as part of the waste form and package factor and the site selection factor. The waste form and package factor, f_{wi}, which is in units of m^3/yr and stands for the source term, and the site selection factor, f_{si}, which is in units of yr/m^3 and stands for the migration reduction factor, are considered below.

It should be noted that equation (4-62), which was developed for routine and activated metal waste streams, is also applicable to source waste streams with some minor modifications. For source waste streams the interaction factor I_i would have units of m^{-3}, while C_{in}, the activity of the source, has units of Ci. Variations in this formulation for source waste streams are considered in the appropriate sections below.

It is also important to recognize that due to certain aspects of the disposal facilities and site environments considered in this report, judgement is required by the code user in interpreting results of groundwater migration calculations. The groundwater migration calculations depend upon an assumption that the rate that water can percolate through the disposal cell covers is approximately equal to, if not less than, the rate that water can infiltrate out the bottom of the disposal cells. Otherwise, the percolating water can accumulate within the disposal cells as leachate. This can happen, for example, at a disposal facility located at a humid site having very impermeable soils (or an impermeable liner), if the permeability of the disposal cell covers exceeds that of the soil (or liner). In this case impacts associated with this leachate accumulation would be more appropriate to consider than those associated with groundwater migration. These leachate accumation impacts are addressed in Section 4.4.

4.3.1 Source Term

The source term is represented by the waste form and package factor f_{wi}, which has units of m^3/year, and which denotes the annual volume of contaminated liquid that passes into the aquifer beneath the disposal facility. This factor is given by the formula:

$$f_{wi} = f_i V_w f_c \tag{4-64}$$

where f_i is the fraction of the total disposed waste volume that comprises the (i)th waste stream; V_w is the annual volume of water that percolates through the disposal facility cover and contacts the disposed waste/soil mixture (in m^3); and f_c is the fraction (dimensionless) of the waste radionuclide concentration transferred to the leachate.

The first factor f_i is self-evident; it is the ratio of the volume of the waste stream being considered to the entire volume of waste disposed. For source waste streams equation (4-64) is replaced with the following:

$$f_{wi} = N_{Si} (S_S/S_f) V_w f_c \tag{4-65}$$

where N_{Si} is the total number of the ith source waste stream disposed at the site; S_S is the maximum cross sectional area presented by a single source, and S_f is the effective disposal area of the facility, i.e., the factor f_i is replaced by the fraction of the disposal surface area that comprises the (i)th source waste stream. The parameter V_w remains the same; however, f_c becomes the volume fraction of the waste stream transferred to the leachate (in m^{-3}).

It is assumed in this report that, after the applicable delay time for any special high integrity source container, all the activity in a sealed source is accessible and can be dissolved by the contacting water. Clearly, the fraction of the disposal surface area represented by the waste stream is equal to the ratio of the maximum cross sectional area of the source to the effective surface area of the disposal facility (S_S divided by S_f) multiplied by the total number of sources, N_{Si}. The area S_S has been assumed to be 78 cm^2 (6"x1") in this report (see Section 4.1.3). The effective disposal surface area is given as follows:

$$S_f = \sum_j V_j / (EMP_j \times EFF_j) \qquad (4\text{-}66)$$

where the summation is performed over all waste classes, and where

V_j = volume of waste in the (j)th waste class;
EMP_j = waste emplacement efficiency of the (j)th waste class; and
EFF_j = volumetric disposal efficiency of the (j)th waste class.

A related concept is the total surface area of the disposal facility, S_t, including the spaces between the disposal cells. This area is given by

$$S_t = \sum_j V_j / (EMP_j \times EFF_j \times SEF_j) \qquad (4\text{-}67)$$

where SEF_j is the surface disposal efficiency of the (j)th waste class. In the above equations, EMP_j, EFF_j and SEF_j are supplied by the disposal technology configuration (see Section 2.2.3.2). EMP_j is given by the emplacement index IE, and EFF_j and SEF_j are determined by the disposal technology index ID (see Section 2.2.3). Reference EMP, EFF, and SEF values for the disposal technologies are given in Appendix C. The other two factors in equations 4-64 and 4-65, V_w and f_c, are discussed below.

4.3.1.1 Rainwater Infiltration and Percolation

Clearly, the variable (V_w) is simply the percolating infiltration (p) multiplied by the effective disposal site surface area (S_f), which is the surface area of the disposal cells. (The space between the disposal cells, if any, is not included). The effective disposal site surface area is given by equation (4-66).

There may be several different techniques for calculating the parameter (p) (also called PERC in several references). One of these methods is usually called the water-balance technique (Refs. 15 and 16). The water-balance technique yields percolation components of about 74 mm/yr of water for the northeast site, 180 mm/yr of water for the southeast site, 50 mm/yr of water for the midwest site, and 1 mm/yr of water for the southwest site. These values are applicable to surface conditions at the site prior to construction and operation of a disposal facility. The volume of water percolating in this case will be denoted by V1.

For cases where there exists special disposal facility covers containing low permeabiity layers, and where the integrity of the covers can be assumed, the percolation component may be determined by the Darcy velocity of the least permeable stratum between the waste and the atmosphere (Ref. 17). The Darcy velocity of a material, with hydraulic conductivity (K) in units of m/yr and unit hydraulic gradient, is equal to K m^3/m^2-yr. This number, however, is modified by the fraction of each year during which there is at least 0.01 inch of precipitation (Ref. 2). Therefore, in this latter case, (p) will be calculated from the following equation:

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$$p = K\ (w/365) \qquad (4\text{-}68)$$

where (K) is the hydraulic conductivity of the least permeable layer covering the waste, and (w) is the mean annual number of days with 0.01 inch or more of rainfall. Assuming that a permeability of 3×10^{-7} cm/sec (about 0.3 ft/yr) is applicable for the least permeable stratum of the disposal facility cover, assuming appropriate values for w for each of the reference sites (Ref. 2), and assuming that the ground is frozen for 3 months each year in the northeast site, this yields estimated percolation components of 27 mm, 30 mm, 25 mm, and 0 mm (1 mm assumed), respectively, for the northeast, southeast, midwest, and southwest sites. The volume of water percolating in this case will be denoted by V2.

The specific percolation value used in the calculations depends upon the particular disposal configuration considered, specifically the disposal technology index ID, the utilization index IU, the cover index IC, the compaction index IX, the backfill index IB, and the topmost waste index IT (if IC is equal to 0). A decision matrix on the use of a particular percolation value is presented in Table 4-6.

TABLE 4-6 . Percolation Value Decision Table

Waste Cover	Compaction	Percolation Value IU=1	IU>1
Regular	Regular	2xV1	V1
	Moderate	1.5xV1	V1
	Extreme	V1	V1
Improved	Regular	2xV1	V2
	Moderate	2xV2	V2
	Extreme	V2	V2

If the IU index is equal to 1, then either the waste stream is unstable or it is disposed with unstable wastes. In this case, both the IC index and the IX index affect the decision. In either case, IC=0 for a waste class indicates that the waste is disposed beneath another waste class. In this case, percolation is determined by the IC value corresponding to that for the particular waste class specified by the topmost waste index, IT. If the IU index is greater than 1, then the waste is stable and is disposed with stable wastes. Waste disposed using reinforced concrete bunkers or disposed using a grout backfill is also considered to be stable. In these cases, V1 is used if the IC index is 1 (regular cover), and V2 is used if the IC index is 2 (improved cover).

It should also be noted that percolation values less than V1 are assumed to be only applicable until the end of the active institutional control period. This is because an assumed limit to the length of institutional control implies a limit to maintenance activities at a disposal facility which would control burrowing animals or deep-rooted vegetation. This, coupled with the potential for inadvertant human intrusion, further means that methods used to reduce percolation into disposal cells will suffer reductions in effectiveness over time. This effect is considered in Section 4.3.3.2 of this report.

One final point of note concerns the relative conservatism of the above assumptions. For example, no reduction of percolation is assumed either for waste grouting (other than the fact that all grouted wastes are assumed to be stable) or for use of disposal technologies involving heavy use of concrete such as waste repackaged in concrete overpacks. These two effects, however, could easily be incorporated into the above percolation value decision table through the use of the backfill index IB.

4.3.1.2 Radionuclide Waste-to-Leachate Transfer

The factor f_c represents the fraction of the activity that is transferred from the waste to the leachate. There are two different approaches to calculation of f_c depending on whether or not the waste is a source waste stream. These are considered below.

For routine wastes f_c may be calculated using the following formula:

$$f_c = f_L M_o t_c \qquad (4\text{-}69)$$

where the factor f_L is the waste leachability/accessibility multiplier, M_o is the fraction of a specific radionuclide transferred from disposed waste to trench leachate due to contact of water at continuous full saturation, and t_c is a correction factor applied to M_o. These are discussed below.

The factor f_L is the multiplier due to the relative leachability/accessibility of radioactivity in the waste stream. For activated metal wastes, it is assumed to be the water accessibility multiplier f_{AW} discussed in Section 4.1.3. For other waste streams, f_L depends on whether the waste is solidified or whether waste containing chelating agents and chemicals have been disposed in a segregated manner. In this case, the multiplier f_L is used as discussed in Section 4.1.3 under index I7.

One additional modifier of the value of f_L is dependent on the backfill index IB (see Section 2.2.3). For backfill index values of 2 (imported sand backfill), and 3 (grout backfill), the multiplier f_L is multiplied by a factor of 0.1. This accounts for the cases in which the percolating precipitation will either pass through the disposal cell expeditiously, or it will have great difficulty contacting the wastes.

The factor M_o can be estimated by many methods. In this report, the average upper bounds of the leach fraction are estimated assuming that the leachate/waste conditions at the Maxey Flats disposal facility may be used to approximate this bounding fraction. As discussed in Reference 2, radionuclide concentrations in leachate were obtained from leachate samples obtained from several disposal trenches. These concentrations were then ratioed to the concentrations in the waste disposed in the disposal trenches. The ratios obtained from each trench were then averaged over all trenches sampled. The ratios thus obtained are listed in Table 4-7.

TABLE 4-7 . Radionuclide Partition Ratios Between Waste and Leachate

Element	Ratio	Element	Ratio	Element	Ratio
H*	1.15E+0**	Ru	1.15E-1	Ac	4.11E-3
C*	5.76E-3	Ag	1.62E-4	Th	4.11E-3
Na	1.62E-4	Cd	1.48E-2	Pa	4.11E-3
Cl	1.15E-1	Sn	1.62E-4	U*	1.25E-4
Fe	1.48E-2	Sb	1.62E-4	Np	4.67E-4
Co*	1.48E-2	I	1.15E-1	Pu*	4.67E-4
Ni	1.48E-2	Cs*	1.62E-4	Am*	4.11E-4
Sr*	9.86E-3	Eu	1.11E-4	Cm	4.67E-4
Nb	1.11E-2	Pb	4.11E-3	Cf	4.11E-4
Tc	1.15E-1	Ra	4.11E-3		

* These ratios are obtained from Reference 2, and were calculated based on experimental data. The remaining ratios were estimated based on chemical similarities between radionuclides.

** Exponential notation, i.e., 1.15E+0 = 1.15 x 10+0.

The primary rationale for this approach is that under specified chemical conditions there is an upper limit to the solubility of all elements. The above disposal site, because of the presence of organic chemicals and chelating agents and because the waste trenches can be assumed to be at continuous full saturation, may be assumed to represent extreme leachability conditions. In addition, the disposed waste was generally in either an unsolidified form or was combined with very poor quality binders. This estimate also takes into consideration the fraction of the leached radioactivity that may be reversibly adsorbed by the interstitial trench soils. Some researchers believe that use of Maxey Flats estimates represent the best that can be achieved with the available experimental data (Ref. 4).

The use of the factor M_o, however, necessitates a correction factor to take into account the expected transient and partially saturated conditions. This correction factor is approximated through t_c. There appears to exist no experimental information that can be used to quantify this parameter. Two theoretical approaches are discussed below.

It was assumed in the original Part 61 analysis methodology that radionuclide release depended on the contact time between the waste and infiltrating water. Assuming that leaching at partial saturation is proportional to the moisture content, the fraction (t_c) may be expressed as the fraction of a year that the percolation component calculated above takes to pass through a given horizontal plane, i.e.,

$$t_c = p/(nv) \qquad (4\text{-}70)$$

where p is the precipitation (in m/yr) that infiltrates and comes into contact with the waste, n is the waste cell effective porosity, and v is the speed of the percolating water (in m/yr). The waste cell effective porosity can be conservatively assumed to be about 25% (partially compacted soils are likely to have higher porosities resulting in lower contact times). The value of v depends on the interstitial soils; a very conservatively low value of 1 ft/day (corresponding to a permeability of about 1x10-4 cm/sec), an effective porosity of 0.25, and a hydraulic gradient of unity will be assumed in this report. The values for t_c for each of the reference sites are presented in Table 4-8.

TABLE 4-8 . Values of Contact Time Fraction - I

	Volume of Water		
Region	V1	2 V1	V2
Northeast	2.66×10^{-3}	5.39×10^{-3}	1.29×10^{-3}
Southeast	6.47×10^{-3}	1.29×10^{-3}	1.08×10^{-3}
Midwest	1.80×10^{-3}	3.59×10^{-3}	9.00×10^{-4}
Southwest	3.60×10^{-5}	7.19×10^{-5}	3.60×10^{-5}

It should be noted that an increase or decrease in the volume of percolating water affects the contact time linearly, which is further incorporated into the formulation of the source term f_{wi}. Therefore, the source term is a quadratic function of percolation. It should also be noted that the contact time fraction approximation should only be considered reasonable over a relatively small range of percolating water speeds. Use of the contact time fraction also implies that the disposed waste is not in continuous full saturation.

Since the publication of reference 2, however, there has been some comment on whether the use of the contact time as the correction factor is sufficiently conservative. Thus, a second method to approximate t_c has been developed for this report. This method is based on the expected average moisture content of the disposal trenches, and may represent an extremely conservative condition.

Variations in moisture content at each site could range from saturation to a residual saturation where capillary tension prevents further gravity drainage. Naturally occurring residual saturations normally have moisture contents greater than 0.05. Residual saturation at a given site is a function of infiltration, hydraulic conductivity and pore size distribution of waste cell, hydraulic properties of trench cap, hydraulic properties of materials between trench bottom and water table, and depth to water table.

Variations in average moisture contents are expected among reference sites and among types of trench covers. Unfortunately, no reliable data on average waste cell moisture content is available for the reference sites. The southwest site is expected to have lower moisture contents based on lower infiltration and greater depths to the water table. Lower trench cap permeabilities are expected to decrease average moisture contents. Based on these qualitative observations, average moisture content values are assigned by reference site and type of trench cap in Table 4.9.

TABLE 4-9 . Values of Contact Time Fraction - II

	Volume of Water		
Region	V1	2 V1	V2
Northeast	0.3	0.6	0.2
Southeast	0.3	0.6	0.2
Midwest	0.3	0.6	0.2
Southwest	0.2	0.4	0.1

Either of these alternatives can be used with the existing codes. The data on both correction factors t_c (called TSC1 and TSC2 in the codes, respectively) are read from the file ENVIRO.DAT, and averaged using a factor between 0 and 1, TSCW, input by the user, i.e.,

$$t_c = \text{TSC1} \times \text{TSCW} + \text{TSC2} \times (1.0 - \text{TSCW}) \qquad (4\text{-}71)$$

For source waste streams, the factor f_c represents the volume fraction of the waste stream transferred to the leachate, and it is given by:

$$f_c = M_o \, t_c \, / \, V_S \qquad (4\text{-}72)$$

where M_o is the radionuclide partition ratios given in Table 4-7, t_c is the correction factor discussed above (Tables 4-8 and 4-9), and V_S is the volume of each source. V_S is assumed to be 309 cm^3 (see Section 4.1.3). Thus, the applicable equation becomes:

$$f_c = 3.24\times10^3 \, M_o \, t_c \qquad (4\text{-}73)$$

4.3.2 Migration Reduction Factor

The waste form and package factor, as expressed above, yields the total source term (in m^3/yr) that can be expected from a given waste stream, and the product of the radioactive concentration with the source term gives the annual release (in Ci/yr). This source term must be related to the concentrations at the groundwater biota access locations. This relation is expressed through the site selection factor (f_s) having units of yr/m^3.

This factor, which has also been referenced as the "confinement factor" or "reduction factor" (Refs. 2, 18), is the ground water migration analog of the (X/Q) dispersion factor in meteorological diffusion calculations.

This report uses a variation of the migration model utilized in the draft and final Part 61 EIS (Refs. 2, 18, 19). This model assumes that the porous medium consists of an unsaturated and saturated zone, each of which is stationary, homogeneous and isotropic, and the fluid moving through these zones is incompressible and of constant viscosity.

Also very importantly, use of this model requires an assumption that any liners placed beneath the disposal facilities are non-functional, or at least only partly functional. Hazardous waste facilities are required by EPA regulation to install liners, but to date there is little data regarding their overall effectiveness.

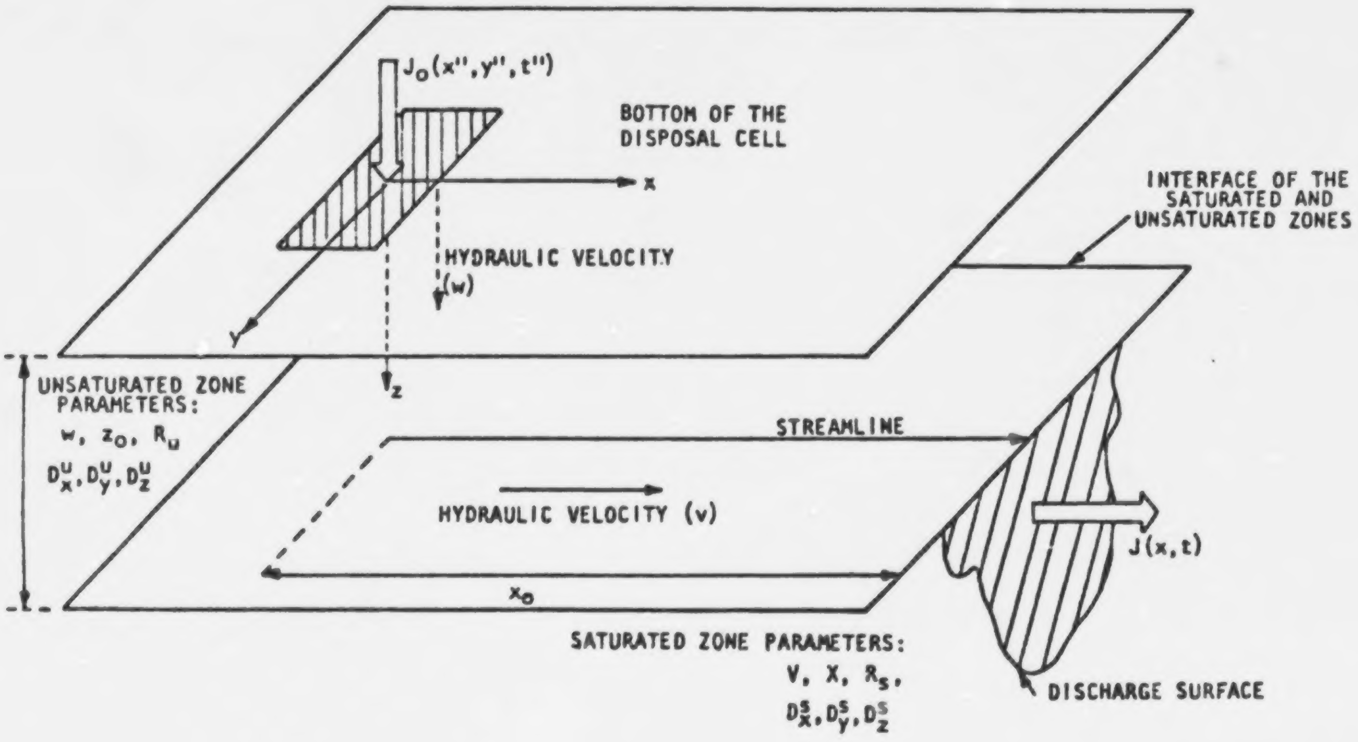
The source term is assumed to be given by J_o (which is equal to f_{wi} multiplied by the waste radionuclide concentrations), whose units are in Ci/yr. The source term is assumed to exist during the source duration time (T). A geometry of the migration problem is shown in Figure 4.6.

The measurable hydrogeological parameters that must be included in a simulation of mass transport are: the geometry of the problem (e.g., the travel distance, x, to a biota access location), the hydraulic velocities of the fluid (e.g., v), the dispersion characteristics of the medium, and the retardation coefficients of the soils through which radionuclides are migrating. The space- and time-averaging of the above parameters, if necessary, may be accomplished in a straightforward manner (see Ref. 2).

It can be shown that the site selection factor is given by (Refs. 2, 18):

$$f_{si} = [r_g r_{ui}/Q] \sum_j r_{tij} \qquad (4\text{-}74)$$

where (Q) is the dilution factor in units of volume/time; the factor r_g is the time-independent reduction factor due to the geometry of the problem (i.e., the spatial relationship of the disposal facility and the biota access location); r_{ui} is the unsaturated zone reduction factor; j denotes


$J_o(x'', y'', t'')$
BOTTOM OF THE DISPOSAL CELL
x
y
z
HYDRAULIC VELOCITY (w)
INTERFACE OF THE SATURATED AND UNSATURATED ZONES
UNSATURATED ZONE PARAMETERS:
w, z_o, R_u
D_x^U, D_y^U, D_z^U
STREAMLINE
HYDRAULIC VELOCITY (v)
$J(x,t)$
x_o
SATURATED ZONE PARAMETERS:
$V, X, R_s,$
D_x^S, D_y^S, D_z^S
DISCHARGE SURFACE

GEOMETRY OF GROUNDWATER MIGRATION

FIGURE 4.6

the longitudinal sectors of the disposal facility shown in Figure 4.5; and r_{tij} is the reduction factor due to migration and radioactive decay which depends on both space and time, including the sectors of the disposal facility and the source duration time (T).

4.3.2.1 Geometric Reduction Factor - r_g

This reduction factor is assumed to be independent of the characteristics of the waste streams. It is also independent of the longitudinal relationship of the disposal facility with the biota access location. Given the generic nature of the analysis methodology in this report, it is conservatively assumed that r_g is equal to its maximum value, unity.

4.3.2.2 Dilution Factor - Q

The dilution factor is independent of the characteristics of the waste stream and the geometrical relationship of the disposal facility with respect to access location. The factor Q may be the pumping rate of a well or the flow rate of a stream.

In this work, the assumed dilution rates are 200,000 m^3/year (about 100 gpm - gallons per minute) for the population well scenario and 4.5 x 10^6 m^3/yr (about 5 cfs - cubic feet per second) for the surface stream scenario. Small farming communities that utilize groundwater for their needs usually have wells that range from 100 gpm to 1000 gpm depending on the population. A stream flow rate of about 5 cfs is selected since a stream with flow rate below this value is very unlikely to be used for human consumption. For example, Rock Lick Creek nearby the Maxey Flats disposal facility has an annual average flow rate of about 7 cfs, but it is not used for human consumption; it is used only for livestock.

For the intruder and boundary well scenarios, Q is given by the assumed total volume of precipitation infiltrating through the disposal faciity area under natural site conditions. In other words, the source term, J_o, calculated in the previous section is diluted by a minimum volume of water (infiltrating through the disposal area and recharging the groundwater) equal to the product of the site percolation (Vl) with the total surface area of the disposal facility, S_t, given by equation (4-67).

There is a lower bound, however, for the value of the dilution volume for the intruder well. Otherwise the above technique would give invalid results for disposal facilities located in regional environments in which the natural percolation is low, e.g., a semi-arid environment. The lower bound dilution rate in this report is taken to be 7700 m^3/year (3.84 gpm), which represents the minimum practical needs of a single person living in a rural area (Ref. 20).

4.3.2.3 Unsaturated Zone Reduction Factor - r_{ui}

In the original Part 61 analysis methodology, the reduction factor for the unsaturated zone was considered as part of the overall migration reduction factor r_{tij}. In this work, however, a separate reduction factor has been developed for the unsaturated zone. This is because for some locations, the unsaturated zone is as important as the saturated zone, if not more so, in affording travel time for radionuclide decay, and it is efficient and desirable to separate the two effects.

In this report, dispersion in the unsaturated zone will be neglected since it introduces severe complications in the calculational algorithm, and does not result in an appreciable change in the amount of radioactivity entering the saturated zone. It will be assumed that the unsaturated zone reduction factor is given by the following equation:

$$r_{ui} = \exp[- \lambda \ R \ x_u/v_u] \qquad (4\text{-}75)$$

where λ is the decay constant of the radionuclide; x_u is the water travel distance between the waste and the saturated zone (in m); v_u is the water travel speed in the unsaturated zone (in m/year); and R is the retardation coefficient of the radionuclide. The parameters assumed for the reference regional site locations are presented in Table 4-10. These parameters are contained in a file called ENVIRO.DAT, and may be altered at the code user's option.

TABLE 4-10 . Reference Groundwater Parameters

	Northeast	Southeast	Midwest	Southwest
Unsaturated Zone				
Speed (m/yr)	0.1	0.2	0.1	0.1
Thickness (m)	0.0	2.0	7.0	28.
Saturated Zone				
Speed (m/yr)	0.1	1.25	0.66	10.
Dispersivity (m)	0.05	0.05	0.05	0.05
Distance to Surface Water (m)	1000.	1000.	2000.	NA*

* NA : Not Applicable

The retardation coefficients (R) that are utilized in the above equation depend on the radionuclide as well as the geochemistry of the soils and the transporting groundwater. They are indicative of the reversible ion exchange capability of the soils and represent the ratio of the radionuclide velocities in the soil to the groundwater velocities. The cation exchange capacity of the soils is a parameter which can be used to estimate the retardation coefficients of the soils, since retardation coefficients are usually linearly dependent on the cation exchange capacity. Five optional sets of retardation coefficients, presented in Table 4-11, may be selected by the user of the code.

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TABLE 4-11 . Assumed Retardation Coefficients[a]

Element[b]	Valence	Set 1	Set 2	Set 3	Set 4	Set 5	BNWL[c]
H	1	1	1	1	1	1	1
C	2,4	10	10	10	10	10	10
Na	1	85	174	353	720	1470	50
Cl	1,5,7	2	3	4	5	7	1
Fe	2,3	630	1,290	2,640	5,400	11,100	3,333
Co	2,3	420	861	1,760	3,600	7,380	333
Ni	2,3	420	861	1,760	3,600	7,380	333
Sr	2	9	18	36	73	147	100
Nb	3,5	1,000	2,160	4,630	10,000	21,600	10,000
Tc	6,7	2	3	4	5	7	1
Ru	3,4,6,8	2	3	4	5	7	--
Ag	1	85	174	353	720	1,470	--
Cd	2	9	18	36	73	147	100
Sn	2,4	85	174	353	720	1,470	1,111
Sb	3,5	9	18	36	73	147	100
I	1,3,5,7	2	3	4	5	7	1
Cs	1	85	174	353	720	1,470	1,000
Eu	2,3	9	18	36	73	147	250
Pb	2,4	840	1,720	3,510	7,200	14,800	16,666
Rn	0	1	1	1	1	1	1
Ra	2	30	61	123	250	508	500
Ac	--	300	609	1,230	2,500	5,080	5,000
Th	4	840	1,720	3,510	7,200	14,800	50,000
Pa	--	840	1,720	3,510	7,200	14,800	16,666
U	4,6	840	1,720	3,510	7,200	14,800	14,286
Np	4,5,6	300	609	1,230	2,500	5,080	100
Pu	3,4,5,6	840	1,720	3,510	7,200	14,800	10,000
Am	3,4,5,6	300	609	1,230	2,500	5,080	10,000
Cm	3	300	609	1,230	2,500	5,080	3,333
Cf	--	300	609	1,230	2,500	5,080	--

(a) Sets 1 and 4 values for H, Fe, Co, Ni, Sr, Tc, I, Cs, Np, Pu, Am, and Cm are obtained from Reference 20; values for other elements are based on chemical similarities and comparative retardations using the BNWL column (Ref. 21). Sets 2 and 3 are obtained as geometric midpoints of sets 1 and 4, and set 5 is similarly calculated.

(b) Coefficients for all other isotopes of the same element are assumed to be the same.

(c) These values are given in Reference 21 for desert soils with a moderate cation exchange of about 5 meq/100g. They have been used as a guide to help fill in missing values.

The clay and mineral content of the soils, in addition to the groundwater chemistry, significantly affects the retardation capability of the soils. The retardation coefficients given in Table 4-11 span the general range of values that are encountered in groundwater migration calculations. The first set is representative of coefficients for sandy soils with low to moderate cation exchange capacities, and is assumed to represent the lower bound of retardation coefficients used in this analysis. The fourth set given is representative of coefficients for clayey soils with moderate to high cation exchange capacities, and is assumed to represent the best conditions that can be routinely achieved. In between these two sets, two other sets have been postulated and have been calculated utilizing the geometric mid-points of sets 1 and 4. A fifth set of coefficients has been also calculated for use in special cases.

Any one of the five sets of retardation coefficients may be selected by the user of the code for a given site. As a reference, however, the 4th set is assumed in this report for the northeast site, the 3rd set for the southeast and the midwest sites, and the 2nd set for the southwest site. Reduced retardation is assumed, however, for waste streams contacting or containing chelating agents and/or organic chemicals. For example, assume that the third set is normally assigned for a particular site. If waste streams containing chelating agents or organic chemicals are disposed in a segregated manner from other waste streams, then the second set of retardation coefficients are assumed for the streams containing the chelating agents and/or organic chemicals and the third set is assumed for the other waste streams. If there is no such segregation, than the reduced set (in this case the second set) is assumed for all waste streams.

4.3.2.4 Saturated Zone Reduction Factor - r_{tij}

This factor depends on the time, after initiation of the scenario, that the exposure is assumed to occur, the duration of groundwater travel between the jth longitudinal sector making up the disposal facility and the biota access location, the retardation capability of the soils (radionuclide dependent), the duration of the assumed source term, and the waste stream characteristics. The longitudinal extent of the disposal facility is considered by dividing the facility into 10 sectors and summing the contributions from each sector (assumed to be equal in size) to obtain the concentrations at the biota access location. In this work, the following formula is used for the migration reduction factor r_{tij}:

$$r_{tij} = [\exp(-\lambda t)/(J x T_i)] \times [F_j(t) - F_j(t-T_i)] \qquad (4\text{-}76)$$

where λ is the decay constant of the radionuclide; t is the time at which the migration reduction factor is applicable; J is the total number of longitudinal sectors the disposal site has been divided into, which is 10 in this work (see Figure 4.5); T_i is the source duration time for the ith waste stream; and j denotes the sector of the disposal site. The function $F_j(t)$ is given by the following formula (see Refs. 18, 19):

$$F_j(t) = 0.5 \times U(t) \times [\mathrm{erfc}(X_-) + \exp(P_j)\ \mathrm{erfc}(X_+)] \qquad (4\text{-}77)$$

$$X_{\pm} = \frac{P_j}{2} \quad \frac{1 \pm t/(Rt_{wj})}{\sqrt{t/(Rt_{wj})}} \qquad (4\text{-}78)$$

where U(t) is the unit impulse function that is zero for a negative argument and is equal to unity otherwise; t_{wj} is the saturated zone water travel time between the disposal sector being considered and the biota access location; P_j is the Peclet number for the distance between the disposal sector and the biota access location; r is the retardation coefficient of the radionuclide; and erfc(x) is the complement of the error function and is given by the formula (Ref. 21):

$$\mathrm{erfc}(x) = 1 - \int_0^x (2/\pi)\ \exp(-t^2)\ dt \qquad (4\text{-}79)$$

The source duration time T_i for the ith waste stream is determined by dividing the total activity in the stream with the annual release fraction which is given by the factor f_{wi} multiplied by the radionuclide concentration. Each year a fixed "slug" of activity is released until the radionuclide inventory is depleted after T_i years. No "exponential" source term, in which the fraction of the initial site inventory released each year is exponentially reduced, is considered.

The groundwater travel times t_{wj} depend on: (1) the distance between the disposal facility sector being considered and the biota acccess location, and (2) the groundwater speeds in the saturated zone. The travel time between the projection of the center of the first sector onto the saturated zone and a biota access location is denoted by t_{w1}. Due to large variability in site locations and sizes, these travel times are calculated internally based on a set of assumed groundwater speeds for the reference environments, the geometry of the disposal site (which is calculated internally based on the waste volumes), and the distance (in m) from the center of the disposal site to the nearest downstream surface water discharge location (see Appendix C for details of calculations).

The assumed reference groundwater speeds are given in Table 4-10. For the southwest site the groundwater speeds and travel times are assumed considering the uniformity of topography and groundwater regimes, and the semi-arid nature of the southwest site with the ensuing low stream densities and high evaporation during periods of high precipitation.

The Peclet number, P_j, is the distance from the disposal sector to the biota access location divided by the longitudinal dispersivity of the medium. These values are also calculated internally in a manner similar to the travel times. The reference longitudinal dispersivities, which are given in Table 4-10, may be altered by modifying the ENVIRO.DAT file.

It may also be pointed out that using groundwater travel times (and the Peclet numbers discussed above) as the primary variables on which the migration analysis is based implicitly allows for a sensitivity analysis. Sites with differing environmental parameters may lead to similar radionuclide concentrations at the access locations. For example, similar results would be obtained if the groundwater velocity is twice as high and the distance to the access location is twice as large, etc.

4.3.3 Special Cases

This section considers two special cases utilized in the groundwater migration calculations: high integrity containers and loss of cover integrity as a function of time.

4.3.3.1 High Integrity Containers

High integrity containers are packages which are designed to preclude contact with water for long periods of time. This is incorporated into the analysis through a delay time which is added to the groundwater travel times for all waste streams packaged in high integrity containers.

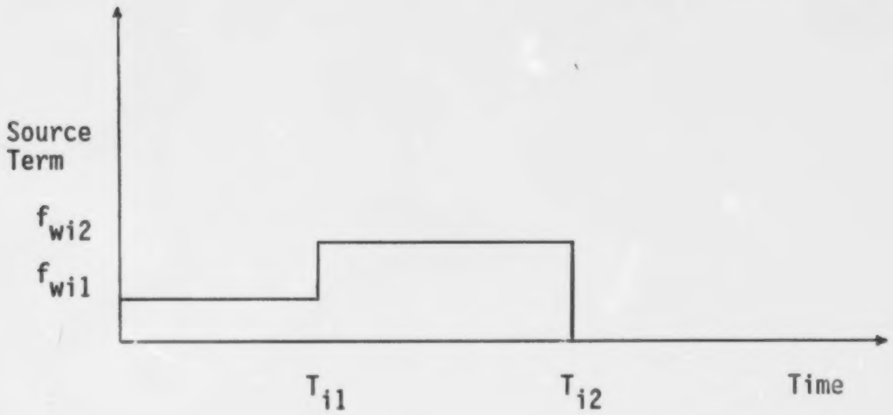
Waste streams contained in high integrity containers are identified by the stability index (I8=3). For source waste streams, the delay time is given by the source delay time, f_{sc}, discussed in Section 4.1.3. For all other waste streams, the delay time is assumed to be 300 years which can be changed by the user by altering a data statement in the code.

4.3.3.2 Loss of Cover Integrity

Although a disposal facility will be designed and operated so that infiltration of rainwater will be minimized, it is possible that sometime after the disposal facility is closed, active institutional controls may breakdown and potential inadvertant intrusion into part of the wastes may occur. As a result, rainwater infiltration may increase. Similarly, a breakdown in institutional controls may lead to intrusion into the waste mass by deep-rooted plants and burrowing animals which also may lead to an increase in rainwater infiltration. This potential increase in infiltration would result in a corresponding increase in the groundwater migration source term. A calculational procedure to account for this time dependent increase in source term is presented below.

For the case of the time dependent source term analysis, two different source magnitudes are considered. The source term is assumed to increase after the end of the active institutional control period as represented by the diagram on the next page.

Two source terms, denoted by (f_{wi1}) and (f_{wi2}), are calculated using equations 4-64 and 4-65. These source terms are used in conjunction with two source duration times denoted by (T_{i1}) and ($T_{i2} - T_{i1}$). The first source term is applicable during a duration time of T_{i1} years, and the second source term is applicable during a duration time of ($T_{i2} - T_{i1}$) years. The second source duration time is calculated by subtracting the



radioactivity that has migrated from the site during the first source duration time from the activity inventory of the site (the area under the above histogram), and dividing the remaining site activity inventory by the second source term. That is, it is calculated by the formula:

$$T_{i2} = T_{i1} + f_{wi1} \times (TDUR - T_{i1}) / f_{wi2} \qquad (4\text{-}80)$$

where TDUR represents the source duration time if f_{wi1} were the source term during the entire period. In other words, TDUR is the duration time for the time independent source term analysis, and TDUR times f_{wi1} times C_n is the entire site inventory of the radionuclide being considered.

For calculational convenience, the source term for this analysis is taken to be equal to f_{wi1} for all times, and the effect of the increased source term after time T_{i1} is incorporated into the factor r_{tij}. The following equation is used to calculate the modified factor r_{tij}:

$$r_{tij} = [\exp(-\lambda t)/(JxTDUR)] \times [F_j(t) - F_j(t-T_{i1}) + (f_{wi2}/f_{wi1} \times [F_j(t-T_{i1}) - f_j(t-T_{i2})] \qquad (4\text{-}81)$$

where $F_j(t)$ is the function defined previously by equation (4-77), and where the other variables are also as previously defined.

This analysis is only considered for disposal cells in which improved covers and operational procedures are used to reduce percolation below that which would occur naturally at the site in the absence of the disposal facility (i.e., percolation less than V1). Ten percent of the disposal cell area is assumed to be disturbed by intruder activities, and the percolation over this area is increased to V1. The remaining 90% of the disposal area is assumed to experience percolation equal to twice the original value.

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For cases where the source is depleted within the active institutional control period (TDUR is less than ICLS+IOBS+IINS), or for cases where the percolation volume at the disposal facility is greater than or equal to V1 (see Section 4.3.1.1), this analysis is ignored.

4.4 Leachate Accumulation Impacts

One of the difficulties faced when modeling groundwater impacts is that the disposal facility design may include installation of disposal cell liners, or the facility may be located in geologic media having low permeabilities. This situation complicates groundwater migration calculations, since percolating water can potentially accumulate in rather than flow through the disposal cells.

This "bathtub" effect has been observed at low-level radioactive waste disposal faciities in the past, and if the water is allowed to continue to accumulate so that the disposal cell fills up with leachate, the disposal cell could overflow, in which case potential public impacts could occur. Alternatively, the disposal site operators could pump the accumulating leachate from the disposal cells, send the leachate through a treatment process, and then release the treated leachate to the environment. The leachate treatment process would also involve potential public impacts, since no treatment process will be fully effective in removing the radioactive constituents from the leachate.

An order of magnitude estimate of potential impacts from leachate treatment and overflow is thus provided. This estimate is adopted from a calculational procedure outlined in Reference 3. Briefly, three scenarios are considered:

1. During operation of the disposal facility, accumulating leachate is continuously removed by site operators, processed in some manner, and released into a nearby stream. The contaminated stream water is consumed and used by an individual, and impacts to the individual are calculated in units of mrem/yr. This scenario is referred to as the "treatment" scenario.
2. Accumulating leachate is allowed to completely fill up the disposal cells to the point that the leachate overflows the disposal cells. The overflowing leachate is carried to a nearby stream. The contaminated stream water is consumed and used by an individual, and impacts to the individual are calculated in units of mrem/yr. This scenario is referred to as the "overflow" scenario.
3. Rather than overflow, the accumulated leachate in the second scenario above is assumed to be processed through an evaporator. Radionuclides released through the evaporator stack impact the surrounding population (in person-mrem/yr). This scenario is referred to as the "evaporation" scenario.

The first scenario could correspond to a situation in which a disposal facility is designed for leachate collection and removal. Example treatment processes could include filtration and ion exchange systems. The

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second scenario could correspond to a situation in which insufficient attention is given during operations to accumulating leachate. The third scenario could correspond to a situation in which remedial actions are taken at a nonoperational facility.

Impacts from the first two scenarios (H_1 and H_2) and the third scenario (P) are thus given as:

$$H_1 = \sum_i \sum_n f_o (A_{in}/Q) f_{1n} \text{ PDCF-7} \tag{4-82}$$

$$H_2 = \sum_i \sum_n f_o (A_{in}/Q) f_{2n} \text{ PDCF-7} \tag{4-83}$$

$$P = \sum_i \sum_n f_o{}' A_{in} f_{rn} \text{ POP PDCF-3} \tag{4-84}$$

with:

$$A_{in} = C_{in} f_i \text{ VL } f_L M_o t_c \tag{4-85}$$

where:

f_o = time delay factor for the first and second scenarios (see below);
$f_o{}'$ = time delay factor for the third scenario (see below);
A_{in} = activity (Ci) contained within the leachate volume (VL) annually generated or processed;
Q = flow rate of surface stream (assumed to be $4.5\text{x}10^6$ m^3/year - see Section 4.3.2.2);
f_{1n} = radionucide release fraction from operational leachate treatment process;
f_{2n} = radionuclide retention in soil during overland flow;
f_{rn} = radionuclide fraction released into the air due to evaporator operation;
POP = population-weighted atmospheric dispersion factor (See Section 3.1.1);
C_{in} = concentration (Ci/m^3) of nth radionuclide in ith waste stream;
f_i = fraction of the total site waste volume that is composed of the ith waste stream;
VL = volume (m^3) of leachate annually released or processed;
f_L = waste-to-leachate correction factor (see below);
M_o = radionuclide-specific leach fraction (see Table 4-7); and
t_c = fraction of a year that the leachate is in contact with the waste.

Parameters requiring additional explanation include f_o, $f_o{}'$, f_{1n}, f_{2n}, f_{rn}, VL, f_L, and t_c. Of these parameters, only f_o, $f_o{}'$, and f_L depend on the waste type, i.e., whether it is routine waste, activated metal, or a source waste stream.

The time delay factors, f_o and f_o', account for radionuclide decay. The first and second scenarios are conservatively assumed to occur during the last year of facility operation. Consequently, this factor is given by the following:

$$f_o = (1/\mathrm{ILFE}) \sum_{k=0}^{\mathrm{ILFE}-1} \exp[-\lambda k] \tag{4-86}$$

where is the radionuclide decay constant (yr-1) and ILFE is the facility life in years, i.e., IEND-IBEG+1. This expression assumes that the waste is delivered to the facility in approximately equal fractions throughout the life of the facility, and each term above accounts for the decay of the waste disposed in a specific year. For the third scenario, the factor f_o' is given by the following:

$$f_o' = f_o \exp[-\lambda \mathrm{TDEL}] \tag{4-87}$$

where TDEL is assumed equal to the closure period, ICLS, i.e., the scenario occurs at the beginning of the observation period.

Clearly, if the waste stream is packaged in a high-integrity container, f_o is set equal to zero. For source waste streams, f_o is assumed to be zero if the source delay time is more than the facility life, i.e., $f_{SC} > \mathrm{ILFE}$.

The radionuclide release fractions for operational leachate treatment, f_{1n}, apply only to the first scenario. These release fractions are radionuclide-specific, and are defined for a given radionuclide as the activity released by the treatment process divided by the total activity input to the treatment process (i.e., the inverse of the decontamination factor for the radionuclide treatment process). Given the very wide range in treatment process that could be implemented, f_{1n} is conservatively assumed to be unity for all radionuclides in this report. The user of the code may optionally program different values of f_{1n} based upon further research and site-specific information.

The factor f_{2n} is also radionuclide-specific, and applies only to the second scenario. It gives, for a given radionuclide, the fraction that is retained in soil during transport from the disposal cell to the stream. Due to a lack of site-specific data, these retention fractions are conservatively assumed to be unity for all radionuclides. The user of the code may optionally substitute alternative values for f_{2n} based on further research and site-specific information.

The radionuclide release fractions for the evaporator, f_{rn}, are assumed based upon information obtained from Reference 24, and upon assumed chemical similarities of particular elements. The values for the pathological incinerator given in Table 3-2 are assumed.

VL is difficult to estimate, but for this report will be assumed to be equal to the total volume of water percolating into the disposal cells per year. This volume is given as the percolation (p) times the effective disposal site surface area (S_f), where S_f is given by equation 4-66. This also gives conservative results, since it corresponds to the last year of facility operation when the disposal facility will be filled to capacity with waste. The percolation (p) is a function of the design and operation of the disposal facility which was discussed in Section 4.3.

For routine waste, the factor f_L expresses the relative leachability of the waste as given in Section 4.1.3. For activated metal waste streams, it is replaced by f_{AW}.

The contact time fraction, t_c, also differs depending upon the scenario. For the first scenario, leachate is assumed to be removed continuously during operation, and so percolating water is assumed to flow through the waste rather than saturate the waste continuously. Thus, t_c for the first scenario is given as a function of percolation as discussed in Section 4.3.1 (Tables 4-8 or 4-9). For the second and third scenarios, the waste is assumed to be in continuous saturated contact with the leachate, and so t_c for these last two scenarios is assumed to be unity.

Finally, for source waste streams the term (C_{in} f_i) in equation 4-85 is replaced by the total activity of the (i)th source waste stream at the disposal site divided by the entire disposal space, and the term f_L is replaced by the term f_{SW}, i.e.:

$$A_{in} = (C_{in} N_S / V_{dis}) \, VL \, f_{SW} \, M_o \, t_c \qquad (4\text{-}88)$$

where N_S is the total number of sources in the (i)th waste stream, and V_{dis} is the total disposal space (disposal volume divided by the emplacement efficiency) given by:

$$V_{dis} = \sum_i \sum_j V_{ij}/EMP_j \qquad (4\text{-}89)$$

where i stands for the waste stream, j stands for the waste class, V_{ij} is the corresponding waste volume, EMP_j is the emplacement efficiency in the (j)th waste class, and the other symbols have been explained before. In simpler terms, the sources are assumed to be distributed within the entire disposal space.

4.5 Exposed Waste Scenarios

In these scenarios, the earthen cover over the disposed waste is assumed to be totally or partially removed, resulting in exposure of the waste and dispersal by wind or water. Two hypothetical mechanisms are considered for removal of the cover: (1) the actions of a potential inadvertant intruder

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(see Section 4.2), and (2) erosion of the cover by wind or surface water. For each removal mechanism, two radiological exposure scenarios are considered. One involves suspension and airborne dispersal of contaminated particulates to the area surrounding the disposal site, leading to exposures to the surrounding population resulting from inhalation, immersion and other pathways. The second scenario involves transport of radionuclides via surface water runoff to an off-site surface water body, leading to exposures to individuals using the contaminated water body. This results in four exposed waste scenarios depending upon the particular cover removal mechanism and radionuclide transport agent considered: intruder-air, intruder-water, erosion-air, and erosion-water.

The following equations are utilized to calculate human exposures resulting from these scenarios. For the water transport and access scenarios:

$$H = \sum_{i} \sum_{n} I_{wat} \, C_{in} \, \mathrm{PDCF\text{-}7} \qquad (4\text{-}90)$$

and, for the air transport and access scenarios:

$$P = \sum_{i} \sum_{n} I_{air} \, C_{in} \, \mathrm{PDCF\text{-}3} \qquad (4\text{-}91)$$

where (n) and (i) denote the radionuclides and the applicable waste streams, respectively, I_{wat} and I_{air} are the interaction factors for water and air given by equation 4-5, C_{in} is the radionuclide concentration or activity in the (i)th waste stream, and PDCF-7 and PDCF-3 are the pathway dose conversion factors discussed in Appendix D. Exposures are calculated for the water transfer scenarios to an individual who is assumed to consume contaminated water from the off-site surface water body, and H represents the 50th year annual dose rate in mrem/yr after 50 years of exposure. For the air transport scenarios, exposures are calculated to the surrounding population, and P represents the total population dose rate in person-mrem/yr after 50 years of exposure.

For the intruder-initiated exposed waste scenarios, the waste/soil mixture involved in the intruder-agriculture scenario is assumed to be involved in the exposed waste scenario as well, i.e., the drilling scenario cuttings plus the 306 m^3 of waste excavated during the intruder-construction scenario. For the erosion-initiated exposed waste scenarios, only waste streams that may lie within 5 m of the earth's surface are considered. The values for the components of the interaction factors (i.e., the transfer factors f_o, f_{di}, f_{wi}, and f_{si} - see Equation 4-5) are presented below.

The time delay factor (f_o) is the same for routine, activated metal, and source waste streams. It is defined by:

$$f_o = \exp[\, - \, \lambda \, \mathrm{TDEL}] \qquad (4\text{-}92)$$

where TDEL is the delay time and λ is the decay constant. For the intruder-initiated exposed waste scenarios, TDEL is taken as a minimum to be the period between the cessation of active disposal operations and the end of the active institutional control period, or ICLS+IOBS+IINS.

For the erosion-initiated exposed waste scenarios, the delay time is assumed to be 2000 years for both a regular cover and an improved cover. These values are extremely conservative. Previous estimates on the erosion potential of adequately emplaced cover materials have ranged for 1000 years to 10,000 years to erode 1 meter of soil cover (Ref. 4).

The other three transfer factors are discussed below for routine waste streams. Activated metal and source waste streams are minor modifications of these formulae, and are considered in Section 4.5.3.

The site design factor, f_{di}, for these scenarios will be defined as the fraction of the exposed area that is waste (as opposed to soil). For the intruder-initiated scenarios, it will be taken as the product of the cover mixing efficiency, RMIX (0.25 - see Section 4.2), and the waste volume weighted emplacement efficiencies, i.e.,

$$f_{di} = \sum_j 0.25\ EMP_j\ V_{ij} / V_t \tag{4-93}$$

where EMP_j is the emplacement efficiency in the (j)th waste class, V_{ij} is the disposed volume of the (i)th waste stream in the (j)th class, and V_t is the total waste volume disposed at the facility. For the erosion-initiated scenarios, the cover mixing efficiency is replaced by the surface utilization efficiency, i.e.,

$$f_{di} = \sum_j SEF_j\ EMP_j\ V_{ij} / V_t \tag{4-94}$$

The waste form and package factor, f_{wi}, denotes the total volume of the soil-waste mixture mobilized by the transfer agent per year. In this report, it may be empirically broken down into the following components.

$$f_{wi} = E \times (A_i/d) \tag{4-95}$$

where:

E = soil-waste mixture mobilization rate (in g/m^2-yr) which will be taken to be independent of the waste stream;
A_i = area of the soil-waste mixture (in m^2) that can be identified with the (i)th waste stream; and
d = density of the soil-waste mixture (in g/m^3).

This equation is applicable for both the wind transport scenario and the surface water scenario. Specific values for the above parameters and the site selection factor, f_{si}, are discussed below.

4.5.1 Wind Transport Scenario

For the erosion-initiated scenarios, the airborne mobilization rate (E) is obtained from calculations presented in Section 5.5.3.1 of Reference 3. Based upon average wind speeds at each of the reference sites, E is approximated as 5.53×10^{-4} mg/m^2-sec at the northeast site, 1.54×10^{-5} mg/m^2-sec at the southeast site, 2.05×10^{-3} mg/m^2-sec at the midwest site, and 7.95×10^{-3} mg/m^2-sec at the southwest site. The density of the soil/waste mixture is assumed to be 1.6 g/cm^3.

Since for the erosion-initiated scenario the entire disposal facility is assumed to be exposed to wind transfer, A_i is equal to S_t as given by equation 4-67. It should be noted that all waste classes are considered in the calculation of the areas.

For the intruder-initiated scenarios, calculations are more complicated since there are four different intruder scenarios considered in this report, and each scenario involves different exposed waste areas. The largest exposed waste areas are associated with the intruder-construction and the intruder-agriculture scenario. For the former scenario, a 200 m^2 area is assumed to be exposed for 500 hours out of the year while for the latter scenario, an 1750 m^2 area is assumed to be exposed for an entire year. In addition, the two intrusion scenarios involve dust mobilization initiated by both man-made and natural mechanisms, and the fractions of a year that the two mechanisms are applicable differ for each scenario. This means that a time-weighted average mobilization rate must be calculated for each scenario. Finally, the cover mixing efficiency (RMIX) is considered for the intruder-agriculture scenario but not for the intruder-construction scenario.

Considering the above factors, it may be demonstrated that total off-site releases are similar for either intruder scenario. Thus, the intruder-agriculture scenario is assumed for calculation of off-site impacts. In this case, A_i is multiplied by 1750 m^2 and divided by the effective disposal site surface area, S_f (see equation 4-67). For Class C waste this value is further multiplied by a factor of 0.1.

To calculate the time-weighted dust mobilization rate for the intruder-agriculture scenario, an elevated dust loading is assumed for 100 hours out of the year which accounts for activities such as tilling which could raise significant quantities of dust through the mechanical disturbance of the soil. The remainder of the year (8670 hours) natural wind suspension is assumed to prevail. As detailed in Section 6.4 of Reference 3, the time-weighted average values for E will be assumed to be 2.03×10^{-3} mg/m^2-sec for the northeast site, 2.50×10^{-3} mg/m^2-sec for the southeast site, 4.49×10^{-3} mg/m^2-sec for the midwest site, and 6.84×10^{-2} mg/m^2-sec for the southwest site.

The site selection factor, f_s, is similar for both the intruder- and erosion-initiated scenarios. For the former, it is equal to POP as discussed in Section 3.1. For the latter, POP is multiplied by three to account for population growth during the years between site closure and initiation of the scenario.

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4.5.2 Surface Water Scenario

Based on surface water erosion calculations (see Appendix A of Reference 2), the mobilization rate for both surface water scenarios (i.e., the factor E in equation 4-95) is calculated to be 1.84×10^{-2} g/m^2-year. This factor corresponds to an annual erosion rate of about 0.82 tons/acre. Annual erosion rates vary with the soil properties, vegetation, prior erosion, topography, etc. The annual erosion rate for the Appalachian region for the past 125 million years has been calculated to be 0.75 tons/acre (Ref. 4). The other factors in equation 4-95 remain as defined in Section 4.5.1 for the respective erosion-initiated and intruder-initiated scenarios.

The surface water site selection factor can be estimated by considering the flow rate of a nearby stream assumed to be utilized by a member of the population. In this report, the site selection factor is given by the inverse of twice the value of the dilution factor Q previously utilized to determine groundwater impacts at the surface water biota access location, i.e., 1.11×10^{-7} yr/m^3. This accounts for the increased flow conditions during heavy precipitation and subsequent heavy stream flow rates. This assumption corresponds to dilution of the released radioactivity in a stream with a flow rate of about 10 cubic feet per second, and it is conservative since a stream with a flow rate this low is unlikely to be utilized for human consumption.

4.5.3 Activated Metal and Source Waste Streams

For both of these waste streams, the above formulae are valid with minor modifications. For activated metal waste streams, this modification consists of the use of the factor f_{AD} (see Section 4.1.3) to multiply the interaction factors. For the source waste streams, in addition to the factor f_{SD}, the source density d_S is used as the multiplier. These factors make the units of the interaction factors for source waste streams m^{-3} which is consistent with the use of the activity for C_{in}.

4.6 Operational Scenarios

This section considers non-routine and routine human exposures which may occur during facility operations. Two accident (non-routine) scenarios have been postulated in this report: accident-container and accident-fire. The main purpose of including these accident scenarios is to assist the code user in investigating the potential for improved waste forms to reduce potential operational impacts. These two scenarios are discussed in Sections 4.6.1 and 4.6.2, respectively. Routine occupational exposures are considered in Section 4.6.3.

4.6.1 Accident-Container Scenario

In this hypothetical scenario, an accident is assumed to occur in which a waste package drops from a significant height (e.g., from a sling, hoist, or crane), breaks open, and releases a portion of its contents into the air. This airborne radioactive material may be transported off-site, leading to subsequent human exposures. Radionuclide release may be modelled as a "puff," and the resulting human exposures calculated over a short time period. Workmen near the accident may also be very briefly immersed in the puff of radioactive particles. Potential exposures to off-site individuals and workers would obviously be a strong function of waste form -- i.e., improved, less dispersible waste forms lead to lower potential releases and reduced potential impacts.

In this report, equations are presented (and programmed into the IMPACTS code) to estimate impacts to an off-site individual. This is the approach used for the original Part 61 analysis methodology. However, the user may also be interested in estimating potential impacts to an on-site worker, and so equations which may be used to estimate such impacts are also briefly discussed. These may be programmed into the IMPACTS code at the user's option.

To model impacts, a reference level of particulate release is first considered assuming a very dispersible waste form as applied to a release scenario obtained from waste transportation accident calculations. Then, a correction factor is applied to take into consideration the relative dispersibility of the waste form. Each calculation is performed for a single waste stream.

Off-site Releases

The equation used to estimate exposures to a human located outside the disposal facility site is as follows:

$$H = \sum_n I_{air}\, C_n\, \text{PDCF-1} \qquad (4\text{-}96)$$

where H is the 50-year dose committment in mrem, I_{air} is the interaction factor for airborne release, C_n is the radionuclide concentration in the particular waste stream considered, PDCF-1 is the pathway dose conversion factor discussed in Appendix D, and n denotes the particular radionuclide considered. As usual, the interaction factor is equal to

$$I_{air} = f_o\, f_d\, f_w\, f_s \qquad (4\text{-}2)$$

Since the accident occurs during the operational period, the time delay factor f_o is assumed to be unity. Similarly, the design and operation of the site is assumed not to significantly influence the impacts, and so the design and operation factor f_d is assumed to be unity. The waste form and package factor, f_w, is given by the following:

$$f_w = f_r\, V_q\, f_q\, f_C \qquad (4\text{-}97)$$

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where

f_r = fraction of the package contents released, assuming a very dispersible waste form (assumed to be 10^{-3} - Ref. 25);
V_q = volume of the container (4.81 m^3 - see below);
f_q = frequency of accident (yr^{-1}); and
f_C = operational dispersibility factor (see below).

The factor f_r is estimated based on work performed by the Department of Transportation (DOT) to determine requirements for transport of radioactive material. In this work, an accident of moderate severity is assumed in which 0.1% of the contents of a waste package is released into the air (Ref. 25). This is believed to be conservative as indicated by Table 31 in Reference 25 which lists respirable mass fractions (generally less than 0.1%) for various fine grained and dispersible materials.

The volume of the container, V_q, is conservatively assumed to be that for a 170 ft^3 liner (4.81 m^3). The factor f_q accounts for the annual frequency of an accident of sufficient extent to result in the above radioactive release. The authors are not aware of any published data on the frequency of such accidents at low-level waste disposal facilities. The factor f_q is therefore conservatively assumed to be unity.

The factor f_C is a dimensionless correction factor which accounts for the relative dispersibilities of various waste forms in comparison with an extremely dispersible waste form. This factor is difficult to determine, but is assumed to be given by the multiplier associated with the scatter index ISC (see Section 4.1.3). It is given by the expression:

$$f_C = 10^{-ISC} \tag{4-98}$$

where ISC indicates the accident scatter index. f_C is assumed to be zero for source waste streams which are contained in a high integrity source container such as that described in Appendix B. (It also may be reemphasized that the above equation is primarily used for comparison purposes.) The site selection factor f_s is given by the equation:

$$f_s = (X/Q)\ EDF \tag{4-99}$$

where EDF is the exposure duration factor for the impacted individual which is given as the ratio of the air inhaled in one intake by a person performing light activity (1.25 liters) to the annual individual inhalation volume (8000 m^3); this value is calculated to be 1.56 x 10^{-7} (Ref. 3).

For puff releases, the atmospheric dispersion factor, X/Q, for a ground level release for a person standing in the centerline of the puff is given in Reference 26 as follows:

$$X/Q = [\pi\sqrt{2\pi}\,\sigma_x\,\sigma_y\,\sigma_z]^{-1} \qquad (4\text{-}100)$$

where σ_x, σ_y, and σ_z are the standard deviation factors for the puff release in three dimensions and where $\sigma_y = \sigma_x$. These factors in units of distance indicate the spread and dilution of the plume as a function of distance from the source.

Given the mostly generic nature of this study, the standard deviation factors are given in a conservative manner. σ_x and σ_y are determined for each of the three sites assuming a one second puff and utilizing the average wind speed for each of the sites. The standard deviation in the vertical direction, σ_z, is obtained from Figure 1 of Reference 26. Assuming a conservative stability Class F for the duration of the accident, σ_x is determined to be about 2.2 m for an individual located at 100 meters from the release. The resultant X/Q values (in units of yr/m^3) turn out to be 9.68×10^{-11} for the northeast, 1.40×10^{-10} for the southeast, 6.21×10^{-11} for the midwest, and 4.11×10^{-11} for the southwest environmental locations.

Given that an accident of such severity as that postulated above is projected to occur only once, and the fact that the puff release lasts for significantly less than a year, population impacts are not calculated. For individuals located more than a few hundred meters for the site, dispersion and dilution effects would result in only negligible impacts.

On-Site Releases

The equation used to calculate impacts to a worker is very similar to that for an offsite individual. Based on Reference 3, it may be estimated as follows:

$$H = \sum_n C_n\, f_r\, f_i\, f_C\, V_q\, f_q\, f_x\, \text{PDCF-2} \qquad (4\text{-}101)$$

where PDCF-2 is pathway dose conversion factor no.2 discussed in Appendix D, and all other factors except f_i and f_x are as described above for off-site releases. The factor f_i (like f_r) is based on consideration of estimates performed for transportation safety purposes. It represents the fraction of the material dispersed into the air that is inhaled by an individual. It is taken to be 0.001 (Ref. 3).

The factor f_x accounts for the accident condition. In this report, the usual manner in which impacts from air concentrations of radioactivity are calculated is to first obtain an airborne concentration in Ci/m^3 valid during a year, and then multiply it with the pathway dose conversion factor in units of mrem/yr per Ci/m^3. All pathway dose conversion factors used in this report implicitly include the assumption that the individual inhales a volume of 8000 m^3 of air during the year. However, for accident conditions, the assumption that a maximum of only 10^{-6} of the radioactive material in the package is inhaled preempts the use of an average concentration during the year. Consequently, the pathway dose conversion factor must be multiplied by 1/8000 m^3/yr.

These impacts are assumed to be applicable to only a few site workers, i.e., only for those in the immediate vicinity of the working area. These include general laborers and equipment operators.

4.6.2 Accident-Fire Scenario

In this scenario, it is assumed that a fire accidentally starts in a disposal cell and lasts for approximately two hours. A portion of the radioactivity is released into the air where it is transported off site and leads to subsequent exposures to humans. Potential exposures from this scenario are a strong function of the waste form and facility design and operation. For example, a waste disposal cell in which all of the materials would be composed of compressible material would involve larger releases than a case in which compressible material is mixed with non-combustible waste. Even though most compressible wastes (which are assumed to be combustible) have very low levels of contamination, improvements in the form of the compressible material would involve lower potential releases. For example, compressible material which has been processed by incineration and solidified would involve lower potential releases than compressible waste which has been processed by compaction.

In this report, the accident-fire scenario is used to help assess the effects of improved waste forms and site operational practices on reducing potential exposures from an accident involving an operational fire. Using this scenario, a waste stream or groups of waste streams may be tested separately. The accident-fire scenario is assumed to be possible only if (1) the waste stream being tested is combustible, or (2) the waste stream being tested is mixed during disposal with other waste streams containing combustible material. The equation used to describe off-site exposures to an individual is:

$$H = \sum_n I_{air} C_n \text{ PDCF-1} \qquad (4\text{-}102)$$

where H is the 50-year dose committment in mrem, PDCF-1 is the pathway dose conversion factor discussed in Appendix C, C_n is the radionuclide concentration in the particular waste stream being considered, n stands for the radionuclide, and I_{air} is the interaction factor (dimensionless) for an airborne pathway. I_{air} is given by

$$I_{air} = f_o f_d f_w f_s \qquad (4\text{-}2)$$

In a manner similar to the accident-container scenario, the time delay factor f_o and the site design and operation factor f_d are both assumed to be unity. The waste form and package factor is given by:

$$f_w = V\, f_F \qquad (4\text{-}103)$$

with

$$f_F = f_r\, 20^{-IFL} \qquad (4\text{-}104)$$

where V is the volume of the waste involved in the fire in units of m^3, f_F is the flammability multiplier associated with the flammability index IFL (the second digit of the accident index, I4 - see Section 4.1.3), and f_r is the radionuclide-specific release fraction discussed in Section 4.1.3.

V is estimated to be 100 m^3 based on an estimated annual disposal volume of 50,000 m^3, two disposal cells operating simultaneously, 250 working days per year, and one disposal cell involved in the accident (Ref. 2).

The site selection factor f_s is given as the product of the exposure duration factor for the exposed individual with the atmospheric dispersion factor (X/Q). The exposure duration factor is taken to be 1.90×10^{-5}, which is the ratio of the time spent by the exposed individual in the centerline of the plume (10 minutes) to the number of minutes per year (525,600). It is not reasonable to assume that an individual would stand in the centerline of the plume from the fire for more than 10 minutes.

The atmospheric dispersion factor (X/Q) for an accident lasting from 0 to 8 hours is given by the following equation (Ref. 26):

$$(X/Q) = \exp[-h^2/(2\sigma_z)]/[\pi u\, \sigma_y\, \sigma_z] \qquad (4\text{-}105)$$

where h is the release height (assumed to be zero in this case), u is the wind speed which is specified to be 1 m/sec assuming Pasquill Stability Class F atmospheric conditions, and σ_y and σ_z are as defined previously. Utilizing values given in Reference 26 at 100 m from the fire yields a (X/Q) factor of 3.62×10^{-3} and a value for the site selection factor of 1.83×10^{-9}.

4.6.3 Routine Exposure Scenarios

Two types of routine exposures associated with disposal facility operations are considered in this report. The first type depends primarily on the annual number of transportation vehicles arriving at the site (i.e., shipments), and involves exposures associated with checking in, inspection, decontamination, and checking out the vehicles. The second type of exposures depends primarily on the waste packaging and shipping mode parameters and secondarily on the disposal technology used, and involves the unloading and emplacement of the waste into the disposal cells. These are calculated primarily for comparison purposes and are examined below.

The labor requirements for shipment dependent exposures are discussed in Appendix C, and involve primarily radiation safety technicians, quality assurance technicians, and skilled laborers. Table 4-12 presents a summary of the labor requirements.

TABLE 4-12 . Labor Requirements for Shipments

Personnel	Person-Days Per Shipment
Radiation Safety Technicians	0.15
Quality Assurance Technicians	0.05
Skilled Laborers	0.05

The radiation fields associated with these exposures are assumed to depend on the waste shipped. As detailed in Chapter 3, two types of radiation fields are considered. If the waste transport vehicles are calculated to have a radiation field less than 10 mR/hr, then the actual calculated value is used (equation 3-6); if this calculated value is larger, then it is assumed that the vehicle is shielded to reduce the vehicle radiation level to the Department of Transportation limit of 10 mR/hr. Moreover, in a manner similar to the distance factor (DF) calculated for the occupational exposures during stopovers (see Section 3.2.2), a proximity factor of 0.235 is assumed for these individuals.

The second type of exposures are more complex since they depend not only on the packaging and shipment mode, but also on the labor requirements for different disposal technologies. The labor and equipment requirements for the various disposal technologies considered in this report are detailed in Appendix C as a function of the emplacement mode (random or stacked), and packaging and shipment mode parameters. These estimates are presented in Table 4-13.

For segregated waste, the above personnel requirements are assumed to increase by a factor of 1.1. This accounts for the need to operate more than one disposal cell at once, and the time lost while shifting personnel and equipment. For layered waste disposal, i.e., when Class C or D wastes are disposed mixed with lower classes of waste, the personnel requirements are assumed to increase by an additional factor of 1.01. This accounts for the extra requirements on the quality assurance technician (one person out of the ten person crew assumed for emplacement work) to assure proper emplacement.

Finally, the radiation field associated with emplacement work is assumed to be 0.5 mR/hr based on exposures experienced with operating sites. Basically, the average exposure received by a field worker of approximately 1 rem/year is divided by an assumed average 2000 work-hours/year per worker, thereby yielding 0.5 mR/hr.

TABLE 4-13 . Person-Minutes for Disposal per Container

Care Level and Container	Overpack*	All Trenches		Other Methods Except Repackaging Option	
		Random	Stacked	Random	Stacked
Regular Care:					
Large Box	Van	100	120	100	120
	FB	37	60	37	60
Small Box	Van	8	12	9	13
Drum	Van	3	12	4	13
Small Liner	Van	68	83	68	83
Large Liner	LC	600	720	600	720
Special Care:					
Large Box	ST	150	180	150	180
Small Box	ST	13	20	15	22
	LC	125	150	125	150
Drum	ST	5	12	7	14
	LC	43	88	43	88
	SC	100	156	100	156
Small Liner	SC	300	360	300	360
Large Liner	LC	600	720	600	720
Extreme Care:					
Drum	SC	100	156	100	156
	1D	300	360	300	360
Small Liner	SC	300	360	300	360
Large Liner	LC	750	900	750	900

* See Table 3-11 for overpacks.

CHAPTER 4.0 REFERENCES

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5.0 SCENARIO AND DISPOSAL TECHNOLOGY INTEGRATION

This chapter reviews some of the considerations inherent in the approach of 10 CFR Part 61 to waste classification, discusses the applicability of these considerations to the hypothetical establishment of a new class of waste, namely, Class D waste, and presents a discussion of the calculational implications of establishing disposal area concentration limits in addition to the more conventional volume or mass concentration limits.

Section 5.1 outlines some of the waste classification considerations of 10 CFR Part 61, while Section 5.2 discusses implications of area concentration limitations and minimum disposal depths. Finally, Section 5.3 presents a discussion of calculational procedures for disposal technology parameters.

5.1 Background

There have been a number of studies which have addressed the question of establishing radionuclide concentration limits for waste disposal, and some of these studies are referenced in reference 1. Intrusion scenarios involving direct human contact with the disposed waste have frequently been used as a major part of the calculations. The classification system existing today in the 10 CFR Part 61 rule is based significantly on intrusion, although disposal facility stability, groundwater migration, operational safety, and existing operational practices were also significant considerations.

Several sections of Chapter 4.0 have presented a number of assumptions and calculational formulas for estimating impacts to a potential inadvertent intruder as a complicated function of waste form, disposal design, operational procedures, and time. It is useful to briefly review some of the basic considerations influencing the calculations, and to address the implications of using the intrusion scenarios as an important input to setting generic concentration limits for waste disposal. Following this, a more difficult and complicated question is addressed, namely, calculation of land use and residual radiological and economic impacts assuming implementation of the generic concentration limits.

The extensive use of intrusion scenarios, as opposed to other scenarios such as groundwater scenarios and exposed waste scenarios, in establishing radionuclide concentration limits is believed to be principally based on pragmatic considerations. Impacts from intrusion scenarios are linearly related to the radionuclide concentration in the waste, and also seem to be less influenced by site specific conditions than the other scenarios. Impacts from other scenarios are dependent on the total radionuclide inventory rather than the concentration, and it is difficult to use an inventory limit to justify a concentration limit for a particular waste package. Moreover, radiological impacts from these scenarios are heavily dependent on site-specific conditions.

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There are three principal and one minor inadvertent intrusion scenarios which are mainly dependent upon the radionuclide concentration within the waste. The three principal scenarios include the drilling, construction, and agriculture scenarios, and these are summarized in Tables 5.1 and 5.2 as a function of (1) disposal depth, and (2) waste structural stability, operational procedures, and time.

TABLE 5-1 . Matrix of Intrusion Scenario Applicability as Function of Disposal Depth

	Intrusion Scenarios		
Condition	Drilling	Construction	Agriculture
No minimum depth	yes	yes	yes
Exceeds 5 m	yes	10%	10%
Exceeds 10 m	yes	no	no*

* Considered indirectly; exposures from waste exhumed during the intruder-drilling scenario are considered as part of intruder-agriculture scenario.

TABLE 5-2 . Matrix of Intrusion Scenario Applicability as Function of Structural Stability and Time

	Intrusion Scenarios		
Condition	Drilling	Construction	Agriculture
TDEL<500 yrs*			
Unstable Waste**	yes	yes	yes
Stable Waste	yes	no	no
Grouted Waste	yes	no	no
RC*** Disposal	no	no	no
TDEL>500 yrs			
Unstable Waste	yes	yes	yes
Stable Waste	yes	yes	yes
Grouted Waste	yes	yes	yes
RC Disposal	yes	yes	yes

* TDEL = Time after the end of surveillance period
** Unsegregated stable waste is considered unstable.
*** RC - Reinforced concrete

Application of a particular intruder scenario for a particular waste stream depends upon a positive response in each matrix. For example, the full intruder-construction scenario is applied when no requirements have been established for disposal depth, except when waste is disposed in a stable, segregated manner (or in a reinforced concrete bunker or other disposal method that provides structural stability independent of the waste form), and the time following the facility surveillance period is less than 500 years.

The minor scenario, the intruder-discovery scenario, may be visualized as a variation of the intruder-construction scenario. It is considered for (1) stable wastes which are disposed segregated from unstable waste, (2) wastes disposed in a reinforced concrete facility, or (3) wastes disposed using a grout backfill. When considered, it precludes, by definition, consideration of the intruder-agriculture scenario. It is not considered for wastes disposed at a depth exceeding 5 m, or when the time following the disposal facility surveillance period is greater than or equals 500 years.

Use of the above intrusion scenarios as a means of helping to establish a waste classification system requires that some simplifying assumptions be made. This is because the waste classification limits must be applied generally to a wide variety of wastes and disposal site environmental conditions, and also because the system must be simple enough to be used routinely. Persons performing such calculations have generally conservatively assumed that the entire disposal facility (or at least the volume of waste contacted) was at a given radionuclide concentration, and have determined limits for a particular radionuclide by comparing a calculated dose against an established dose limit.

The classification system eventually adopted in the Part 61 rule is relatively simple to use, and its main concepts are listed in Table 5.3 for each of the three existing waste classes. It considers waste form, operational disposal practice, and waste emplacement depth. Also shown in Table 5.3 are some of the main concepts associated with the Part 60 regulation for disposal of high-level waste into geologic repositories, as well as some postulated concepts that could be associated with a new hypothetical "Class D."

Class D waste has been visualized as a class of waste having requirements for near-surface disposal which lie somewhere between the existing Class C and Part 60 requirements. Basic principles include (1) waste stability, and (2) disposal so that the waste would not be affected by surface events (erosion, housing construction, deep rooted plants). One way to accomplish the latter is postulated to be disposal within the ground with at least 10 m of cover by soil or lower activity waste. If this is the case, then about the only way in which the waste could be contacted is via the intruder-drilling scenario described in Section 4.2.1. That is, impacts from the intruder-drilling scenario can be compared with a specified dose limit and this comparison used to set radionuclide-specific concentration limits.

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TABLE 5-3 . Basic Criteria for Waste Classification

Class A	Minimum waste form requirements Segregated disposal from stable waste. Institutional control for at least 100 years
Class B	Minimum waste form requirements Disposal in a structurally stable manner Institutional control for at least 100 years
Class C	Minimum waste form requirements Disposal in a structurally stable manner Institutional control for at least 100 years Disposal considering inadvertant intrusion (e.g., disposal at 5 m minimum depth)
Class D	Minimum waste form requirements Disposal in a structurally stable manner Institutional control for at least 100 years Disposal in a manner removed from near-surface processes (e.g., disposal at 10 m minimum depth)
High-level waste	Minimum waste form requirements Multiple barriers (engineered and geologic) - containment (300-1000 yrs) - release rate (1 part in 100,000 per year after 1000 years) - groundwater travel time (1000 years to accessible environment) Waste retrievability

One difficulty with this approach is that unlike other intruder scenarios involving direct human contact with the waste (construction and agriculture), impacts from the drilling scenario are governed not only by the radionuclide concentration, but also by the radionuclide quantity. The total radionuclide quantity brought to the surface is of concern, which means that in addition to a limit on the radionuclide concentration, a limit is needed in the thickness of the radioactive column accessed.

A general way to accomplish this is to specify a radionuclide-specific area concentration limit in units of Ci/m^2, i.e., a limit in the total activity of a given radionuclide beneath a given surface area. An area concentration limit could be used by licensees (it is practical, more so than a specific dose limit, for example), and could be easily calculated by dividing the radionuclide Class D concentration limit by the waste stratum thickness used to set the concentration. Some of the practical implications of area concentration limits are discussed in the next section.

5.2 Discussion

A major feature of the proposed updated Part 61 methodology is its flexibility -- the ability to calculate land use and residual economic and radiological impacts as a result of alternative waste classification and disposal requirements. However, the imposition of radionuclide-specific area concentration limits somewhat complicates these calculations.

In this report, concentration limits are specified for four different waste classes (A, B, C, and D). The radiological and economic impacts are then determined for each of 6 different "subclasses": Class A (unstable), Class A (stable), Class B, Class C, Class D1, and Class D2. The impacts from each class may be considered separately. Class A and Class A stable must both meet the Class A criteria, but are separated in order to allow flexibility in determining impacts. Similarly, Class D1 and Class D2 must meet the same classification requirements but are separated so that wastes from particular sources (e.g., routine vs nonroutine) may be considered separately. Classes B, C, and D are required to be disposed in a structurally stable manner. Finally, there is assumed to be minimum disposal depth requirements for waste Classes C and D. Different minimum disposal depths may be considered (by changing a data statement in the codes), although in general a minimum depth of 5 meters is assumed for Class C waste and 10 meters for Class D waste.

This is a relatively simple set of criteria to be manipulated, but the implications of applying the criteria can become complicated, especially if an area concentration limit is considered in addition to volume (or mass) concentration limits.

The first question associated with the area concentration limits relates to the classes of waste that these limits would apply. The answer is clear, when Class D is disposed in a segregated manner from other waste classes; it would apply only to Class D wastes. This does not appear to present a problem since these area concentration limits would likely be derived from the intruder-drilling scenario; they would likely be large, and they would probably not affect a reasonable disposal configuration for Classes A, B, or C.

However, when Class D wastes are disposed mixed with other classes of wastes, the answer is not as clear. Since the drilling bit accesses all the waste classes, it would seem that the area limits should apply to all classes of waste accessed. This is the approach adopted in this report.

As a simple example of the implications of setting area concentration limits, assume that one has 100 m^3 of waste containing 1000 Ci of a single particular radionuclide, and that one wished to dispose of this waste in an auger hole having a cross-sectional area of 5 m^2 and a depth of 30 m (see Figure 5.1). This waste mass is assumed to consist of six separate waste streams, all falling into the same Class D. Were there no area concentration limits, one could dispose of the entire waste mass in the bottom 20 m of the auger hole, leaving 10 m on top of the waste to be backfilled with soil. This is illustrated as the first case in Figure 5.1.

Given the depth of waste disposal (10 m minimum), the only intrusion scenario that is postulated to involve contact with the waste is the intruder-drilling scenario. Impacts to an individual (mrem/yr) are proportional to the average radionuclide concentration within the borehole multiplied by the 20 m waste thickness. One problem, however, with this calculation is that each of the 6 assumed waste streams contains the radionuclide of concern in a different concentration. Individual waste packages of a given waste stream may be located anywhere within the disposal stratum, and so the impacts associated with the scenario will vary depending upon the particular combination of waste packages being drilled through. One method of handling this dilemma would be to perform a complex risk assessment which considers, for example, the probabilities of contacting particular waste streams. This would be an extremely complicated task, and clearly not worth the effort, especially when one considers that intrusion is a hypothetical event to begin with.

A more reasonable approach, and one adopted in this report, is to assume that any waste package may be in any location within the disposal cell without preference (also see Section 2.3). An "average" dose rate is then calculated based on the dose rate for each individual waste stream (assuming the entire disposal stratum is filled with the waste stream) multiplied by its fractional volume. This does not give a precise indication of impacts, but a volume average. That is:

$$H_j = \sum_i (V_{ij}/V_j)\, H_{ij} \qquad (5\text{-}1)$$

where

H_j = average dose rate for waste class (j);

V_{ij} = volume of waste stream (i) in waste class (j);

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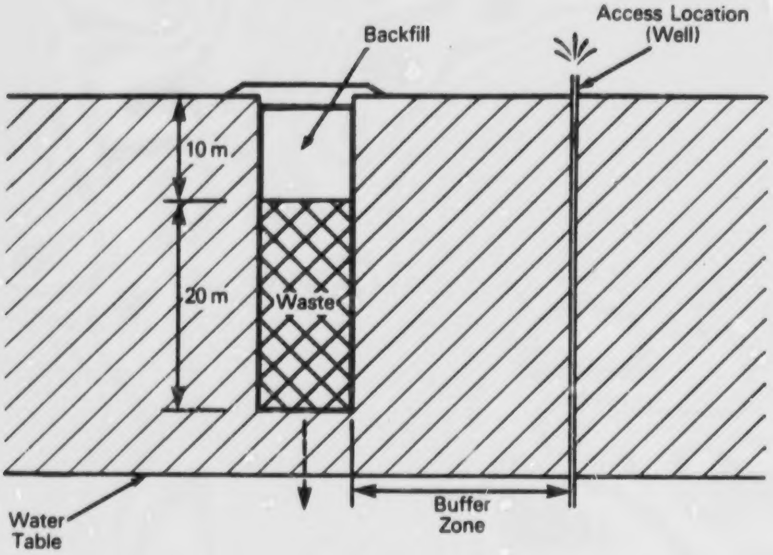
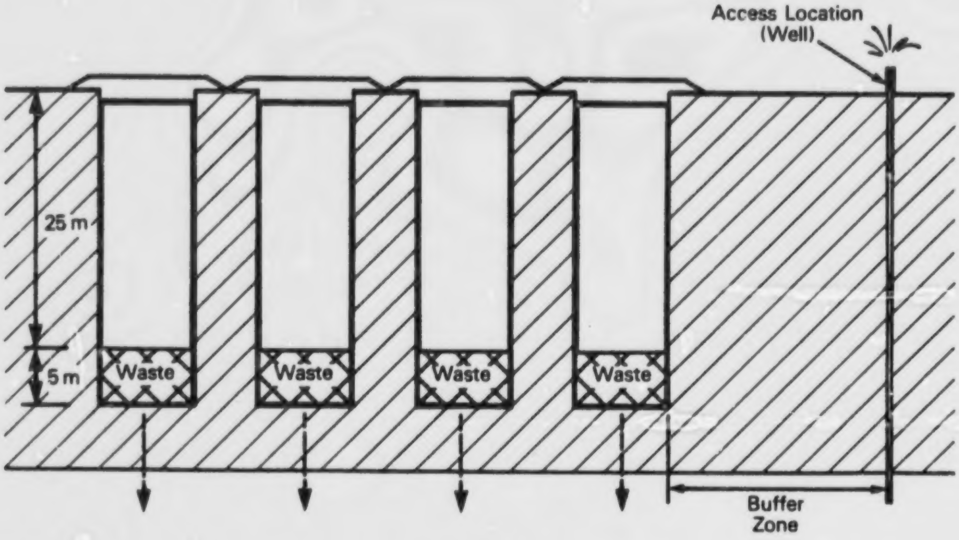

Backfill
Access Location (Well)
10 m
20 m
Waste
Water Table
Buffer Zone

Case 1


Access Location (Well)
25 m
5 m
Waste
Waste
Waste
Waste
Buffer Zone

Case 2

Figure 5.1 Example Area Concentration Limitation Scenario

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V_j = total waste volume in waste class (j); and

H_{ij} = dose rate for an individual waste stream (i) in waste class (j).

The disposal thickness in this case (DTK - see Section 2.2.3) is 20 m for each of the six hypothetical waste streams.

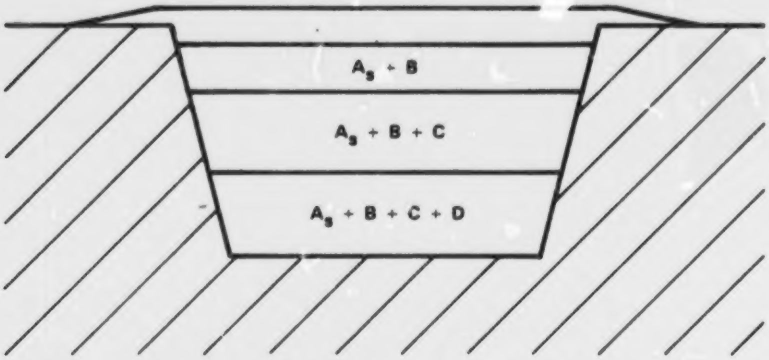
Now assume that there is an area concentration limit of 50 Ci/m^2. This means that no more than 5 m^2 x 50 Ci/m^2 = 250 Ci may be disposed in a single auger hole. Instead of one auger hole, one would need to construct 4 auger holes. In each hole, 25 m^3 of waste would be placed on the bottom 5 m of each auger hole. The remaining 25 m in each auger hole would be backfilled. The resulting disposal cell configuration is illustrated as the second case in Figure 5.1.

The resulting differences in radiological, economic, and other impact measures are significant. For example, four times as much land would be required. However, the potential impacts to an inadvertent intruder from the drilling scenario would be reduced by a factor of four. (The average dose rate, H_j, would be calculated in the same manner as in Case 1 except that DTK would be 5 m rather than 20 m.)

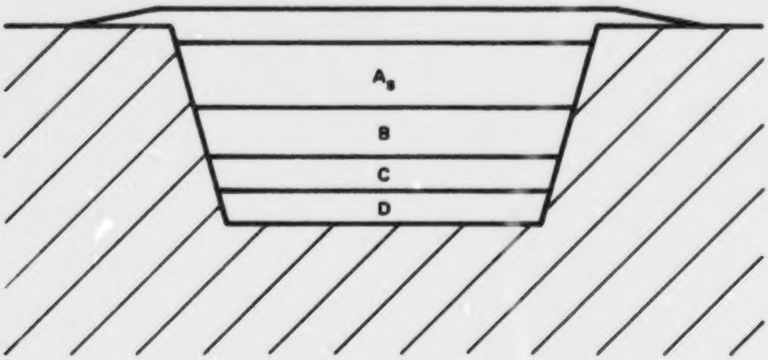
Groundwater impacts would be likely lowered and disposal costs raised, although neither would be a linear multiple of four. In terms of groundwater impacts, assume that the same radionuclide quantity is released from the disposal facility in Case 1 as in Case 2. However, the radioactivity would take longer to reach the biota access location in Case 2 than in Case 1, since significant quantities of radioactivity would travel greater distances. In the case of disposal costs, most of the administration and capital costs would be similar for both cases. Operational costs would clearly be higher in Case 2 (e.g., 4 times as many auger holes to construct and cap, 10 times as much backfill), although the difference would be less than a factor of 4.

Now consider an even more complicated situation in which there is more than one waste class and one radionuclide disposed per disposal cell. For example, consider a large disposal trench 14 m deep and located in an arid environment. Four different classes of stabilized waste are disposed in the trench: Class A stable, Class B, Class C, and Class D. Disposal requirements specify that Class C waste must be disposed having at least 5 m of cover of soil or lower activity waste, while Class D waste must be disposed having at least 10 m of such cover. The disposal geometry is illustrated in Figure 5.2. Waste classes A stable and B can be found anywhere in the trench while Class C waste is only found in the bottom 9 m, and Class D waste is only found in the bottom 5 m.

In principle, a volume-averaged dose rate (i.e., equation 5-1) can be determined for each of the four waste clases in a manner similar to the above example. One would determine impacts for each waste stream as a function of time and disposal depth, and calculate and sum the volume-weighted results. Possible problems with this approach include:


$A_s + B$
$A_s + B + C$
$A_s + B + C + D$

Actual



Conceptual

Figure 5.2 Hypothetical Disposal Cell Containing

Several Classes of Waste

1. Four separate waste classes are involved, and waste streams from some waste classes may be found anywhere in the trench while waste streams from other waste classes will be more localized.
2. The area concentration limit must be applied over all wastes in the trench, which is a mixture of several waste streams in four separate classes.
3. The depth restrictions for Class C and D wastes mean that less disposal space is available within a given trench than for other waste classes. Depending upon the relative waste volumes within each class, there could be situations in which there is insufficient disposal space within a trench for a given class, even though the total volume over all waste classes is less than or equal to the available trench volume.
4. Although not a problem in the above example, there could be situations in which the available disposal depth within the selected disposal technology is not sufficient for the requirements of a given class. For example, the available disposal depth may be only 6 m while the minimum depth requirement for Class D waste is 10 m.

The possible problem with available disposal space can be illustrated using the following example which assumes disposal of two waste classes (Classes B and C) using a single disposal technology. Assume that the total available disposal volume in a disposal cell is 100 m^3, and that depth requirements dictate that Class C waste must be disposed in the bottom fifth of the disposal cell. Thus, only 20 m^3 of Class C waste can be disposed per disposal cell. There are no depth requirements for Class B waste; thus, up to 100 m^3 of Class B waste could be disposed (assuming no Class C waste). Now further assume that there is 1000 m^3 of waste to be disposed given two alternative cases: Case 1: 95% B, 5% C and Case 2: 50% B, 50% C. The number of disposal cells and average waste distribution per disposal cell is given in Table 5-4.

TABLE 5-4 . Two Hypothetical Disposal Cases

Consideration	Case 1	Case 2
Waste volumes	950 m^3 B, 50 m^3 C	500 m^3 B, 500 m^3C
Theoretical number of disposal cells, with no depth consideration (N1)	10	10
Theoretical number of disposal cells, each class considered independently (N2)*	9.5 (B) 2.5 (C)	5 (B) 25 (C)
Actual number of disposal cells	10	25
Average waste volume per disposal cell (m^3)	95 (B) 5 (C)	20 (B) 20 (C)
Total per Disposal Cell	100 m^3	40 m^3

*Disposal depth requirements are considered.

As illustrated, there is no problem with waste disposal for Case 1. The available Class C disposal space exceeds the amount that actually needs to be disposed, and so the excess space can be filled with Class B waste. In Case 2, however, 25 disposal cells are required merely to dispose of the Class C waste.

The procedure for determining disposal cell requirements can be generalized. First, the theoretical disposal cell requirements are determined for each waste class. The maximum value over all waste classes, N2, is compared against the theoretical number of disposal cells that would result if there were no disposal depth restrictions, N1. The actual number of disposal cells is then taken as the larger of either N1 or N2. The average waste volume per class and disposal cell is then taken as the class volume divided by the actual number of disposal cells.

The question of compliance with the area concentration limit can be determined by adopting a relatively simple approximation which is commensurate with the hypothetical nature of the intrusion event. As illustrated in Figure 2, each waste class may be idealized as occupying a specific strata of specific thickness within the disposal trench. Each class strata is assumed to be situated in the trench in order - that is, Class A stable in the top stratum, Class B under Class A stable, Class C under Class B, etc.

Within each waste class, the total inventory (Q) of each radionuclide is determined. This is divided by the surface area of the disposal stratum, and then divided by the area concentration limit. The resulting sum is totaled for all waste classes. To comply with the area concentration limits, the total would be less than or equal to unity. That is, the following inequality would have to be satisfied:

$$T_2 + T_3 + T_4 + T_5 = T < 1 \qquad (5\text{-}2)$$

$$T_j = \sum_n (Q_{nj}/A_j)/AL_n = \sum_{i,n} (Q_{nij}/A_j)/AL_n \qquad (5\text{-}3)$$

$$Q_{nij} = C_{nij} V_{ij} \qquad (5\text{-}4)$$

where (n) denotes the radionuclide, (j) denotes the waste class (there are 4 in this example), (i) denotes the waste stream, and where

Q_{nj} = inventory of radionuclide (n) in waste class (j);

A_j = surface area measured at the top of waste class (j);

AL_n = area concentration limit for radionuclide (n);

Q_{nij} = inventory of radionuclide (n) in waste stream (i) and waste class (j);

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C_{nij} = concentration of radionuclide (n) in waste stream (i) and waste class (j); and
V_{ij} = volume of waste stream (i) in waste class (j).

If the sum of the T value exceeds unity, the waste must be distributed through more disposal cells than 1. In this case, the thickness for each waste class in the disposal cell would be reduced by a factor of T. This would result in the new T being equal to 1.

Performing the above calculations requires that the thicknesses of the class stratums, disposal depths (distances to top of class stratum) and disposal cell volumetric disposal efficiencies be calculated and tracked, as well as the surface areas of each waste class stratum. As a corollary, the plug backfill volumes (V_{bf}) and the number of disposal cells (N3) must be tracked to determine disposal cell costs accurately. (As discussed in Appendix C, waste backfill refers to the backfill between the waste packages, while plug backfill is the material between the top of the uppermost layer of waste packages and the bottom of the disposal cell cover.)

5.3 Calculational Procedures

This section presents procedures for calculation of impacts from disposal of wastes within the framework outlined above and incorporating arbitrary area concentration limits. Disposal technology parameters affected by this recalculation are presented in Table 5-5.

Using the disposal technology indices listed in Table 2-16, it must be initially determined for each waste class whether the class is disposed separately or mixed with any other waste class. Calculational procedures for separate and mixed disposal are different. Criteria for separate disposal are addressed first.

5.3.1 Waste Disposed Separately

The first step is to retrieve the reference values for EMP_r, EFF_r, SEF_r, DPT_r, and DTK_r for the particular disposal technology, and d_{min} for the particular waste class. There are two calculational variations in this section: vertical walls and sloped walls. These cases are illustrated in Figure 5.3, and are addressed separately.

5.3.1.1 Vertical Walls

Step 1 : Retrieve the empty disposal cell volume, V_e, and the cross sectional area, A_{cs}, for the particular disposal technology (see Table 5-5). Cross sectional area is used for disposal technologies featuring vertical side walls (e.g., concrete methods, augers).

Step 2 : Check to see if there is enough available depth in the selected disposal technology to satisfy the minimum depth requirement, (d_{min}). If d_{min} is less than (DTK_r + DPT_r), proceed with the calculation. Otherwise, there is not enough depth to meet the d_{min} requirement. Stop and print an error message that the depth criterion is not met.

TABLE 5-5 . Affected Disposal Technology Parameters

Sloped Wall Technologies	ID	Empty Volume V_e	Size (m) (a)*	Size (m) (b)*	EFF	SEF	DPT	DTK
Ref. trench, humid	1	38,326	30	180	6.23	.88	2.0	6.7
Small trench, humid	2	2,763	10	60	3.88	.69	2.0	4.7
Large trench, arid	3	65,539	30	180	11.4	.88	2.0	13.0
Small trench, arid	4	3,533	10	60	5.25	.69	2.0	7.0
Slit trench, humid	5	1,209	4	60	4.73	.47	2.0	4.7
Slit trench, arid	6	1,227	4	60	4.82	.47	2.0	5.0

Vertical Wall Technologies	ID	Empty Volume V_e	Surface Area-m^2	EFF	SEF	DPT	DTK
Auger, humid**	7	1612.	282.7	4.70	.079	2.0	4.7
Auger, arid**	8	8484.	282.7	29.0	.079	2.0	29.0
Con. trench, humid	9	5033.	860.4	5.7	.44	2.0	5.7
Con. trench, arid	10	5033.	860.4	5.7	.44	2.0	5.7
Con. slit, humid***	11	1316.	225.	5.7	.17	2.0	5.7
Con. slit, arid***	12	1316.	225.	5.7	.17	2.0	5.7
Repackaging, humid	13	28,690	4250.	6.75	.68	2.0	6.75
Repackaging, arid	14	28,690	4250.	6.75	.68	2.0	6.75

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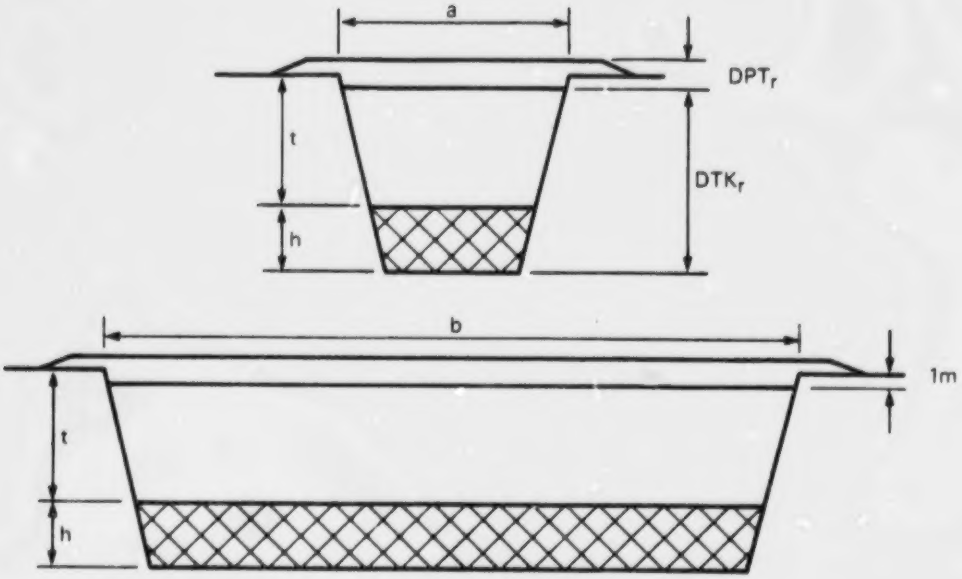
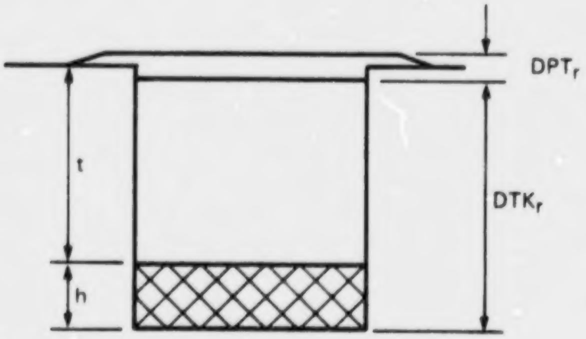
* See Figure 5.3.
** Per group of 40 auger holes
*** Per group of 6 concrete slit trenches


a
DPT$_r$
t
DTK$_r$
h
b
1m
t
h

Sloped Walls


DPT$_r$
t
DTK$_r$
h

Vertical Walls

Figure 5.3. Vertical and Sloped Wall Geometries

Step 3 : Determine a test value for the waste stratum thickness within the disposal cell (h), i.e., $h = (DPT_r + DTK_r) - d_{min}$. However, if d_{min} is less than or equal to 2 m, then $h = DTK_r$.

Step 4 : Check to see if the radionuclide area concentration corresponding to the test h satisfies the area concentration limit (AL_n) using equations 5-2 and 5-3, and the following area:

$$A_j = V_j/(h\ EMP_j) \qquad (5\text{-}5)$$

where V_j is the total volume in waste class (j).

Step 5 : If T is less than or equal to 1, then the area concentration limit criteria are satisfied. Otherwise, the thickness of the waste stratum must be adjusted; in this case, h is is replaced by $h' = h/T$.

Step 6 : Once h' is calculated, then the effective EFF_e, SEF_e, DPT_e, DTK_e can be obtained. (Note that if T is less than or equal to 1, h' is equal to h, and these four parameters are not recalculated.) For cost calculations, the total number of disposal cells (N3) and the plug backfill volume (V_{bf}) must also be known. These parameters are given by:

$$EFF_e = DTK_e = h' \qquad (5\text{-}6)$$

$$SEF_e = SEF_r \qquad (5\text{-}7)$$

$$DPT_e = DTK_r + DPT_r - h' \qquad (5\text{-}8)$$

$$N3 = V_j/(h'\ A_{cs}\ EMP_j) \qquad (5\text{-}9)$$

$$V_{bf} = A_{cs}\ (DTK_r - h') \qquad (5\text{-}10)$$

Step 7 : Calculate impacts. Note that DPT_e is important for determining which intruder scenario is applicable. For example, if $DPT_e > 10$ m, the drilling scenario is considered, along with the agriculture scenario where the source term consists of the wastes exhumed during the drilling scenario. If $0 < DPT_e < 5$ m, either the discovery or the construction and agriculture scenarios are used, along with the drilling scenario. If $5 < DPT_e < 10$ m, the drilling scenario is included along with 10 % of the construction and agriculture scenarios.

5.3.1.2 Sloped Walls

Step 1 : Retrieve the empty disposal cell volume (V_e) and the surface dimensions a and b, for disposal technologies in which the walls slope in at a 1:4 ratio (see Table 5-5).

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Step 2 : Check to see if there is enough available depth in the selected disposal technology to satisfy the minimum depth requirement, (d_{min}). If d_{min} is less than (DTK_r + DPT_r), proceed with the calculation. Otherwise, there is not enough depth to meet the d_{min} requirement. Stop and print an error message that the depth criterion is not met.

Step 3 : Determine a test value for the waste stratum thickness within the disposal cell (h), i.e., $h = (DPT_r + DTK_r) - d_{min}$. However, if d_{min} is less than or equal to 2 m, then $h = DTK_r$.

Step 4 : Check to see if the radionuclide area concentration corresponding to the test h satisfies the area concentration limit (AL_n) using equations 5-2 and 5-3. In this case, calculation of the cross-sectional area is more difficult since it grows smaller with increasing depth. For purposes of estimating impacts of alternative area concentration limits, A_j is conservatively assumed to be calculated using the minimum cross-sectional area of the disposal cell. It is given by:

$$A_j = V_j\ [a-(DTK_r+1)/2]\ [b-(DTK_r+1)/2]/[V_e'\ EMP_j] \qquad (5\text{-}11)$$

$$V_e' = V_e - t\ [ab - (a+b)t/4 + t^2/12] \qquad (5\text{-}12)$$

where V_e' is the available disposal cell volume associated with the waste thickness h, and where $t = DTK_r + 1 - h$.

Step 5 : If T is less than or equal to 1, then the area concentration limit criteria are satisfied. Otherwise, the thickness of the waste stratum must be adjusted; in this case, h is is replaced by $h' = h/T$.

Step 6 : Once h' is calculated, then the effective EFF_e, SEF_e, DPT_e, DTK_e, the total number of disposal cells (N3), and the plug backfill volume (V_{bf}) can be calculated. Using the value $t = DTK_r + 1 - h'$, these parameters are given by:

$$EFF_e = V_e'/[(a-t/2)(b-t/2)] \qquad (5\text{-}13)$$

$$SEF_e = SEF_r(a-1/2)(b-1/2)/[(a-t/2)(b-t/2)] \qquad (5\text{-}14)$$

$$DPT_e = DTK_r + DPT_r - h' \qquad (5\text{-}15)$$

$$DTK_e = h' \qquad (5\text{-}16)$$

$$N3 = V_j/(V_e'\ EMP_j) \qquad (5\text{-}17)$$

$$V_{bf} = t\ [ab - (a+b)t/4 - t^2/12] - [ab - (a+b)/4 - 1/12] \qquad (5\text{-}18)$$

$$V_e' = V_e - t\ [ab - (a+b)t/4 + t^2/12] \qquad (5\text{-}19)$$

Step 7 : Calculate impacts. Note that DPT_e is important for determining which intruder scenario is applicable. For example, if $DPT_e > 10$ m, the drilling scenario is considered, along with the agriculture scenario where the source term consists of the wastes exhumed during the drilling scenario. If $0 < DPT_e < 5$ m, either the discovery or the construction and agriculture scenarios are used, along with the drilling scenario. If $5 < DPT_e < 10$ m, the drilling scenario is considered along with 10 % of the construction and agriculture scenarios.

5.3.2 Mixed Disposal

The first step is to identify the waste classes which share the same disposal cells, to retrieve the reference values for EMP, EFF_r, SEF_r, DPT_r, DTK_r for the particular disposal technology (these parameters are assumed to be independent of the waste class), and to retrieve d_{min} for each waste class. Note that EMP must have the same value for each waste class in a single disposal cell. Again, there are two calculational variations in this section: vertical walls and sloped walls. These are addressed separately.

5.3.2.1 Vertical Walls

Step 1 : Retrieve V_e plus A_{cs}.

Step 2 : Check to see if there is available depth in the selected disposal technology to satisfy the d_{min} requirements. If $d_{minj} > (DTK_r + DPT_r)$ for any waste class, then there is not enough depth to meet d_{min} requirements. Stop and print an error message that the depth criteria are not met. If $d_{minj} < (DTK_r + DPT_r)$ for all waste clases, proceed with the calculations.

Step 3 : Determine the number of disposal cells, and waste class thicknesses, in the absence of an area concentration limit criterion. This is the same as determining which waste class, if any, dominates disposal cell requirements. First determine the theoretical number of disposal cells assuming no depth restrictions (N1), plus the theoretical number of disposal cells for each class ($N2_j$), incorporating depth restrictions and considering each class independently:

$$N1 = \sum_j V_j/[EFF_r \, A_{cs} \, EMP] \qquad (5\text{-}20)$$

$$N2_j = V_j/[A_{cs} \, (DPT_r + DTK_r - d_{minj}) \, EMP] \qquad (5\text{-}21)$$

The maximum value of either N1 or $N2_j$ equals the total number of disposal cells (absent criteria for area concentration limits), or N3. The average waste volume per class per disposal cell, V_{cj}, is then given by:

$$V_{cj} = V_j/N3 \qquad (5\text{-}22)$$

The total average waste volume per disposal cell, V_{ca}, is the sum of V_{cj} over j. Each waste class is assumed to be stratified in successive layers within each disposal cell with the highest class on the bottom. A total waste thickness, h, must then be calculated as an overall average over the disposal cells:

$$h = \sum_j h_j = \sum_j V_j/A_{cs} \tag{5-23}$$

Step 4 : Check to see if the radionuclide area concentration corresponding to the test h satisfies the area concentration limit using equations 5-2 and 5-3. The areas, A_j , are independent of waste class and are given by:

$$A_j = V/(h\ EMP) \tag{5-24}$$

where V is the total waste volume over all classes mixed together in the disposal cells.

Step 5 : If T is less then or equal to 1, then the area concentration criteria is satisfied. If T > 1, then h must be adjusted. h is replaced by h', and the thickness of each waste class stratum h'_j is calculated as follows:

$$h' = h/T \tag{5-25}$$

$$h'_j = (V_j/V)\ h' \tag{5-26}$$

Step 6 : Finally calculate EFF_e, SEF_e, DPT_{ej}, DTK_{ej}, the total number of disposal cells (N3), and the plug backfill volume (V_{bf}):

$$EFF_e = h' \tag{5-27}$$

$$SEF_e = SEF_r \tag{5-28}$$

$$DPT_{ej} = DTK_r + DPT_r - \sum_{k=j}^{6} h'_k \tag{5-29}$$

$$DTK_{ej} = h'_j \tag{5-30}$$

$$N3 = V/(h'\ A_{cs} \cdot EMP) \tag{5-31}$$

$$V_{bf} = A_{cs}\ (DTK_r - h') \tag{5-32}$$

Step 7 : Perform calculations. This approach assures accuracy in determining the location of each waste stream within the disposal cell as a function of depth, performing intruder impact calculations, and performing calculations involving area considerations (e.g. groundwater).

5.3.2.2 Sloped Walls

Step 1 : Retrieve V_e plus a and b.

Step 2 : Check to see if there is available depth in the selected disposal technology to satisfy the d_{min} requirements. If $d_{minj} > (DTK_r + DPT_r)$ for any waste class, then there is not enough depth to meet the d_{min} requirements. Stop and print an error message that the depth criteria are not met. If $d_{minj} < (DTK_r + DPT_r)$ for all waste classes, proceed with the calculations.

Step 3 : Determine the number of disposal cells, and waste class thicknesses, in the absence of an area concentration limit criteria. This is the same as determining which waste class, if any, dominates disposal cell requirements. First determine the theoretical number of disposal cells assuming no depth restrictions (N1), and the theoretical number of disposal cells for each class ($N2_j$), incorporating depth restrictions and considering each class independently:

$$N1 = \sum_j V_j/[EFF_r\ (a-1/2)\ (b-1/2)\ EMP] \tag{5-33}$$

$$N2_j = V_j/[V_e'\ EMP] \tag{5-34}$$

$$V_e' = V_e - t\ [ab - (a+b)t/4 + t^2/12] \tag{5-35}$$

with $t = d_{min} - 1$.

The maximum value of either N1 or $N2_j$ equals the total number of disposal cells (absent criteria for area concentration limits), or N3. The average waste volume per class per disposal cell, V_{cj}, is given by:

$$V_{cj} = V_j/N3 \tag{5-36}$$

The total average waste volume per disposal cell, V_{ca}, is the sum of V_{cj} over j. Each waste class is assumed to be stratified in successive layers within each disposal cell with the highest class on the bottom. A total waste thickness, h, must then be calculated as an overall average over the disposal cells. In this case, the situation is more complicated, since the cross sectional area of the disposal cell changes with depth. One has a choice of solving a cubic equation, or using an "average" cross-sectional area (e.g., measured halfway down the disposal cell). For simplicity, the second approach is adopted, i.e.,

$$A_u = (a - DTK_r/2)\ (b - DTK_r/2) \tag{5-37}$$

$$h = V_{ca}/(A_u \ EMP) \qquad (5\text{-}38)$$

Step 4 : Check to see if the radionuclide area concentration corresponding to the test h satisfies the area concentration limit using equations 5-2 and 5-3, and using the area given by:

$$A_j = A_u \ V/(V_e' \ EMP) \qquad (5\text{-}39)$$

$$V_e' = V_e - t \ [ab - (a+b)t/4 + t^2/12] \qquad (5\text{-}40)$$

where t = DTKr + 1 - h.

Step 5 : If T is less than or equal to 1, then the area concentration criteria is satisfied. If T > 1, then h must be adjusted. h is replaced by h' and the thicknesses of each waste class stratum, h'_j, are calculated as follows:

$$h' = h/T \qquad (5\text{-}41)$$

$$h'_j = (V_j/V) \ h' \qquad (5\text{-}42)$$

The second equation is again an approximation. A more precise way to do it would be to start with the highest waste class -- i.e., that waste class which can be envisioned as occupying the bottom strata. Having calculated the bottom dimensions and the reference DTK for the disposal technology of concern, one could solve a cubic equation for h_1. Once h_1 is known, the top dimensions of the slab can be calculated. These become the bottom dimensions of the next slab above the bottom slab and h_2 is determined by solving another cubic equation. The process is repeated for all waste classes. This is deemed to be unnecessarily complicated and the above approximation is adopted.

Step 6 and Step 7 are identical with those in Section 5.3.2.1.

CHAPTER 5.0 REFERENCES

(1) U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Draft Environmental Impact Statement on 10 CFR Part 61: Licensing Requirements for Land Disposal of Radioactive Waste," USNRC Report NUREG-0782, September 1981.

6.0 DISPOSAL COST CALCULATIONS

This chapter outlines the components that contribute to the calculation of disposal costs and presents a summary of unit costs. These unit costs can be altered by the code user without affecting the cost algorithm embodied in the code ECONOMY. Further details of the cost calculations can be found ad nauseam in Appendix C.

Most of the considerations that affect land use impacts have been outlined in Chapter 5.0. Any remaining considerations are discussed in the "Land" cost component within Section 6.1. Otherwise, this chapter is devoted exclusively to the calculation of disposal costs.

A summary of the disposal technologies and parameters considered in this report is presented as Table 6-1.

TABLE 6-1 . Basic Disposal Technology Parameters

Disposal Technology				Volumes - m^3					
Regular	ID	Location	Volume Basis	Empty Cell	Maximum Waste	EFF	SEF	DPT	DTK
Large Trench	1	Humid	each	38,326	24,370	6.23	.88	2.0	6.7
Small Trench	2	Humid	each	2,763	1,640	3.88	.69	2.0	4.7
Large Trench	3	Arid	each	65,539	45,140	11.4	.88	2.0	13.0
Small Trench	4	Arid	each	3,533	2,210	5.25	.69	2.0	7.0
Slit Trench	5	Humid	each	1,209	739	4.73	.47	2.0	4.7
Slit Trench	6	Arid	each	1,227	752	4.82	.47	2.0	5.0
Auger	7	Humid	40 grp	1,612	997	4.70	.079	2.0	4.7
Auger	8	Arid	40 grp	8,484	6,150	29.0	.079	2.0	29.0
Concrete									
Trench	9	Humid	each	5,033	4,900	5.7	.44	2.0	5.7
Trench	10	Arid	each	5,033	4,900	5.7	.44	2.0	5.7
Slit Trench	11	Humid	6 grp	1,316	937	5.7	.17	2.0	5.7
Slit Trench	12	Arid	6 grp	1,316	937	5.7	.17	2.0	5.7
Repackaging	13	Humid	each	28,690	12,500	6.75	.68	2.0	6.75
Repackaging	14	Arid	each	28,690	12,500	6.75	.68	2.0	6.75

As stated previously, any one of these disposal technologies may be used to dispose any one of the waste classes. An extremely detailed approach to land use and disposal cost calculations is indicated. Thus, an effort has been made in this report to accurately quantify individual components that contribute to disposal cost. This effort consists of separating the disposal facility life into five periods: preoperational, operational, closure, surveillance and institutional control. Within each period all distinct cost components are considered and quantified. A simple present value analysis is presented in Appendix C which integrates all these costs into a few meaningful comparative values. In addition, total annual costs for closure, surveillance, and institutional control may be optionally overridden by the code user to account for possible accidents or contingencies. The five periods of the disposal facility life are individually considered below.

6.1 Preoperational Period

Cost components that contribute to the preoperational period, which is assumed to last five years, are detailed in Table 6-2.

TABLE 6-2 . Preoperational Period Cost Components

Component	Variation
Land	Total surface area
Licensing	Two options
Administration	Constant
Startup overhead	Volume and design
Heavy equipment	Volume and design
Light equipment	Number of personnel
Land Development	Total surface area
Buildings	Volume and design
Utilities	Constant
Engineering and design	% of some components
Contingency	% of all of the above

As shown, several of the components depend on the total surface area of the facility, several depend on waste volume and design mix, others depend on the personnel required, and several are constant. These are briefly considered below.

Land - The land area associated with the disposal facility is divided into four components: A_a - administration area (assumed to be 9.1 acres); A_d - disposal area (calculated using the algorithms presented in Chapter 5); A_b - buffer zone area which is determined using the disposal area and a user input value for the buffer zone width; and A_c - a contingency area assumed to be 30% of A_d plus A_b (the sum of the disposal and buffer areas is also designated as the operational area). An average cost of $2000/acre is assumed. These costs are assumed to be distributed between the five preoperational years as follows: 20%, 2%, 2%, 2%, and 74%.

Licensing - This component is further subdivided into eleven items: site screening and licensing, site characterization, other studies, application preparation, preparation of procedures and site manuals, environmental report preparation, NRC licensing fees, other permits, legal fees, environmental monitoring, and public outreach.

Each of these items are considered in detail in Appendix C, and apportioned between the five assumed preoperational years. Assumed costs of the licensing component (in $1,000) are given in Table 6-3.

TABLE 6-3 . Licensing Costs for Preoperational Period

Option	Year 1	Year 2	Year 3	Year 4	Year 5
Routine	1,150	1,891	392	1,043	350
Extensive	1,150	1,957	457	1,111	350

The difference between the options arises from the assumed cost for "other studies" in the above list (all other items in the list are assumed to be constant). For disposal technologies involving underground engineered concrete structures, additional studies (e.g., seismic studies) are assumed to be performed.

Administration - These costs are assumed to be a constant $646,500/year, and are applied to each of the five preoperational years.

Startup Costs - This component covers the startup overhead costs incurred during the last year of the operational period. It includes costs such as personnel salaries, relocation and travel expenses, and training. It is assumed to depend upon the number of personnel employed during the first year of operation. The number of personnel, however, in a manner similar to heavy equipment requirements, depends on the waste volumes, design mix, and operational alternatives. It is assumed that startup costs are 50% of the salary and benefit costs during the first year of operation.

Heavy Equipment - This cost component is one of the largest contributors to the preoperational costs. Heavy equipment requirements of each distinct disposal technology and operational alternative have been quantified in Appendix C. Thus, the mix and number of heavy equipment are calculated after specification of the waste volumes and design, and operational alternatives. All the heavy equipment is assumed to be purchased during the final year of preoperational period. The heavy equipment considered, their unit costs, and their replacement schedules are given in Tables 6-4 and 6-5.

Requirements for most heavy equipment pieces are estimated as follows. The total number of equipment-days is determined for each piece of equipment as a function of waste volume and disposal design; this is then divided by the product of the equipment use factor (80%) and the number of working days per year (250 days). Requirements for some pieces of equipment, listed in Table 6-5, are based solely on waste volume.

TABLE 6-4 . Heavy Equipment Costs and Replacement Schedules

Equipment	Cost ($1000)	Schedule (years)	Equipment	Cost ($1000)	Schedule (years)
Basic Equipment			Optional Equipment		
Bulldozer	200	10	Auger Rig	400	10
Front end loader	100	10	Stemming unit	4	10
Dump truck	40	10	Paving machine	90	10
Pan scraper	60	10	Tandem roller	35	10
Motor grader	100	10	Compactor	80	10
Backhoe	100	10	Forklift, large	40	10
40-ton crane	150	10	Farm tractor	20	10[b]
100-ton crane	500	10	Hand tamper	1	5
Forklift, small	30	10	Cement truck	45	10
Water truck	55	10	1 CY cem. bucket	5	--[b]
Pickup truck	10	5[a]	Cem. pump&pipes	100	--
4WD truck	15	5[a]	Yard tractor	120	10[a]
Sedan	12	5[a]	Flatbed trailer	20	10[a]
Accessories	100	10[b]	Air monitors	1	--[c]

(a) See Table 6-5; (b) 1 only; (c) See environmental monitoring.

TABLE 6-5 . Equipment Dependent on Waste Volume

Equipment	Annual Volume (m^3/year)				
	2,000	5,000	20,000	35,000	50,000
Standard					
Pickup truck	2	2	3	4	5
4WD vehicle	2	2	3	4	5
Sedan	1	1	1	2	2
Yard tractor	0	0	1	1	1
Flatbed trailer	0	0	1	2	3
Repack Option - Add					
Yard tractor	1	1	0	1	1
Flatbed trailer	1	1	1	1	2

Light Equipment - This cost component includes minor equipment such as furnishings for the administrative and health physics/security buildings, survey meters, radiation detection equipment, typewriters, portable fire extinguishers, etc. Light equipment is assumed to cost $3,500 per site

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employee, is purchased during the last year of the preoperational period, and is replaced every 10 years.

Land Development - This component includes items such as land preparation, access roads, parking areas, on-site roads, fencing, and lighting. These costs are applied in accordance with the following schedule: 0, 10%, 0, 10%, and 80% for the five preoperational years, respectively.

These costs depend on land area and perimeter length. Land preparation is assumed to cost $1450/acre as applied to the sum of the administrative, disposal, and buffer areas. Access roads are assumed to cost $100,000. Costs for parking areas are assumed to depend on the total number of site personnel; 80 m^2 is assumed to be required per person at a rate of $5/$m^2$.

On-site roads, fencing, and lighting all depend on the perimeter length of the operational area (sum of the disposal and the buffer areas). Onsite roads are assumed to cost $50/m, and are 10 m wide. Fencing is calculated as the product of a unit cost of $9.14/m multiplied by the perimeter of the operational area. Lighting is calculated in a similar manner at a cost of $1700/pole installed every 30 m at the perimeter of the site.

In order to calculate these costs, the width of the buffer zone (B_W, in m, which is input by user), and the length (D_L) and the width (D_W) of the disposal area (both in m and internally calculated) are needed. The disposal area (A_d, in acres) is assumed to be in the shape of a golden rectangle for purposes of calculating its length and width. The following equations are applicable:

$$D_L = 1.618\ D_W = 80.92 \sqrt{A_d} \qquad (6\text{-}1)$$

$$P_L = 2(D_L + D_W + 4\ B_W) \qquad (6\text{-}2)$$

$$\text{On-site road cost} = \$50 \times (P_L + 2D_L + 4\ B_W) \qquad (6\text{-}3)$$

$$\text{Fencing costs} = \$9.14 \times P_L \qquad (6\text{-}4)$$

$$\text{Lighting costs} = \$1700 \times P_L/30. \qquad (6\text{-}5)$$

Buildings - This cost component, assumed to be incurred during the last year of the preoperational period, depends on both the number of personnel and the annual waste volume disposed. Six basic and two optional buildings are considered in this report. These buildings, their unit costs, dependency, and cost calculational methods are presented in Table 6-6.

Utilities - This cost component is also assumed to be incurred during the last year of the preoperational period. It includes water, telephone, and electricity installation. It is assumed to be constant at $200,000.

Engineering and Design - This cost component is applied to the last three cost items, namely, land development, buildings, and utilities. It is assumed to be 10% of the total cost of these items.

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TABLE 6-6 . Building Cost Calculations

Basic Buildings	Unit Cost	Dependency	Relationship
Administration	\$463/m^2	Adm/Support Personnel - N1	12.5 N1 + 350 m^2
Health Physics/ Security	\$596/m^2	Guards and Workers - N2	12.5 N2 + 200 m^2 (minimum 300 m^2)
Waste Activities	\$371,000	Constant	
Warehouse	\$331/m^2	All Personnel - N3	6 N3 - 10 m^2 (minimum 200 m^2)
Storage Shed	\$ 10,600	Constant	
Garage	\$331/m^2	All Vehicles and Equipment	210 m^2 for 0-13 420 m^2 for 14-39 630 m^2 for 40+
Optional Buildings			
Waste Processing and Repackaging	---	Annual Waste Volume - V (m^3)	\$600,000+\$20V
Cement Plant	\$160,000	Constant	

Contingency - This cost component is applied at a rate of 20% to all of the above items.

6.2 Operational Period

The cost components contributing to the operational period are presented in Table 6-7, and are discussed below.

TABLE 6-7 . Operational Period Cost Components

Cost Component	Variation
Salaries	Volume and design mix
Disposal cell materials	Volume and design mix
Environmental monitoring	Waste volume
Personnel training and monitoring	Personnel number
Heavy equipment operating expenses	Waste volume
QA and compliance testing	Volume and design mix
Other costs	Constant/waste volume

6.2.1 Salaries

This cost component is assumed to depend upon the number of personnel which is assumed to depend on the waste volumes, design mix, and operational alternatives. The job categories and salaries of the personnel considered in this report are presented in Table 6-8.

TABLE 6-8 . Typical Personnel and Salaries

Senior Staff	Salary ($1000)	Support Staff	Salary ($1000)
Site Manager	50*	Junior Engineer	24*
Executive Secretary	15	Waste Shipment Scheduler	20
Radiation Safety Officer	35*	Billing/Accounting Pers.	15
Assistant Site Manager	40*	Security Personnel	18*
Foreman	28*	Secretarial Personnel	13
Operations Manager	38*		
QA & Safety Supervisor	30*	Workers	
Site Engineer	35*		
Office Manager	38	Quality Assurance Tech	25*
Security Chief	25*	Radiation Safety Tech	25*
Librarian (Records)	15	Heavy Equipment Operators	25*
Customer Service Coord.	24	Skilled Laborers	25*
Contracts Coordinator	24	Unskilled Laborers	15*
Personnel Manager	32*	Surveyor	50*
Regulatory Affairs Manager	35*		

* Indicates badged personnel

The first two groups of personnel, administration and support, are assumed to be dependent on the annual waste volume. Total salaries plus benefits (30% of salaries) assumed for these personnel are given in Table 6-9 as a function of annual waste volume.

TABLE 6-9 . Administrative/Support Personnel
Number and Costs per Year

Item	Annual Volume (m^3/year)				
	2,000	5,000	20,000	35,000	50,000
Total Number	11	13	20	29	36
Badged Number	8	9	12	16	18
Costs ($1,000)	351	391.3	570.7	869.7	1,014

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The third group of personnel, called workers, are calculated based upon the annual waste volume, design mix, and operational practices. Workers, consisting of quality assurance technicians (QA techs), radiation safety technicians (radtechs), heavy equipment operators (HE operators), skilled laborers (S. laborers), unskilled laborers (laborers), and surveyors, are discussed below.

The requirements for workers are difficult to estimate given the large variation in disposal technologies considered in this report, and for which little U.S. experience is available. The approach taken is to identify the principal factors contributing to personnel estimates, and then to estimate person-day requirements for each job category based on disposal technology and annual waste volume disposed. The total number of workers per job category is then determined by dividing the person-day totals by the product of the assumed worker efficiency (75%) and the number of personnel working days per year (220 days). The principal factors assumed to be contributing to personnel estimates are given in Table 6-10.

TABLE 6-10 . Components of Personnel Requirements

Disposal cell construction	Disposal cell covering
Vehicle check in and out	Cement plant operation
Waste processing	Facility maintenance and surveys
Waste emplacement	Environmental monitoring
Waste backfilling	

Disposal cell construction requirements are specified in Appendix C for each disposal technology in terms of person- and machine-days. Thus, first the number of disposal cells required is determined (see Chapter 5), then the unit rates are multiplied with these estimates and summed over all waste classes.

Vehicle check in and out personnel requirements are a function of the annual number of waste shipments which are calculated using the procedures outlined in Chapter 3. Each waste shipment is assumed to require the person-days specified in Table 6-11.

TABLE 6-11 . Vehicle Check In and Out Requirements

Personnel	Person-day/shipment
Radtechs	0.15
QA techs	0.05
S. laborer	0.05

Waste processing is required only in the event that the repackaged disposal option is used at the site (ID = 13 or 14). In this case, there will be a waste process and repack building (WPRB). The personnel requirements for the WPRB are given in Table 6-12.

TABLE 6-12 . Waste Process and Repack Requirements

Occupation	Salary ($1,000)	Annual Waste Volume (m^3/yr x 1000) <10	10<20	20<30	30<40	40<50	>50
WPRB Foreman	28	1	1	1	1	1	1
S. Laborer	25	1	1	1	2	2	2
Laborer	15	1	1	2	2	2	2
QA Tech	25	2	2	3	3	4	5
Radtech	25	2	3	3	4	5	6

Waste emplacement requirements are determined similarly to those for vehicle check in and out. That is, the time requirements for a standard crew given in Table 6-13 are estimated for each container and handling care level.

TABLE 6-13 . Standard Emplacement Crew

Person	Number
Radtech	2
QA tech	1
Heavy equipment operator	2
Skilled laborer	2
Unskilled laborer	3

For the repackaged disposal option, a standard 50 man-minutes per container (bloc) is assumed for waste emplacement. For other technologies, calculations are performed using the person- and machine-minutes specified in Table 4-13 for each different type of container. However, these values are increased by 10% if there is waste segregation. In addition, QA tech requirements are increased by 10% if there is layering, i.e., preferential placement of particular waste classes beneath other waste classes. The assumed fractional equipment mix varies depending on the disposal technology and is given in Table 6-14.

Waste backfilling depends on two different volumes: waste backfill (WB) and plug backfill (PB). Both of these volumes are affected by the waste emplacement method, i.e., whether it is random or stacked. The volume WB is given by the following:

TABLE 6-14 . Machine Requirements for Waste Emplacement (machine-minute fractional distribution)

Equipment	All Trenches	Others Except Repack Option	Repackaging Option
40-ton Crane	0.2	0.67	0.25
100-ton Crane	0.2	0.33	0.25
Large Fork Truck	0.2		0.50
Small Fork Truck	0.4		

$$WB = V\ (1\text{-}EMP)/EMP \qquad (6\text{-}6)$$

where V is the waste volume disposed, and EMP is the emplacement efficiency. The second volume, PB, is calculated using the calculational procedures developed in Chapter 5.

Different unit requirements are estimated for two major operational alternatives: backfill material (soil, sand, or grout), and compaction (none, moderate, and extreme). For soil or sand backfill with minimum and moderate compaction, Tables C-15 and C-16 of Appendix C are used to estimate person- and machine-day requirements. For extreme compaction with these backfill materials, estimates given in Table C-17 of Appendix C are added. (Backfill material costs are considered in Section 6.2.2 -- Disposal Cell Materials.)

For grout backfill, two different unit requirements exist depending on the manner in which the grout must be placed in a disposal cell. For disposal cells having large physical dimensions (Option 1), grout is assumed to be pumped in place using delivery pipes mounted on booms. Option 2 denotes a lack of need for a separate pumping unit (humid auger, slit trench, or concrete slit trench). Requirements for these options are specified in Table 6-15.

Disposal cell covering requirements must be specified for each different disposal technology and for two major cover alternatives: reference, and improved. Unit person- and machine-day requirements for these options are presented in Tables C-18 and C-19 of Appendix C.

Person- and machine-day requirements for cement plant operation, based on an assumed grout or cement production of 50 m^3, are given in Table 6-16.

Facility maintenance and survey requirements depend on the total volume of waste disposed at the facility up to that point in time, i.e., the cumulative volume. These personnel- and machine-day requirements are considered in two groups: (1) maintenance of structures and grounds, plus radiation surveys which are relatively independent of the waste class and site

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TABLE 6-15 . Person- and Machine-Day Requirements For Grout Backfill (Day/1000 m^3)

Personnel	Option 1	Option 2
S. Laborer	8.72	8.72
Laborer	26.2	17.4
Radtech	8.72	8.72
QA tech	0.436	0.261
HE operator	8.72	--
Equipment		
Cement mixer	8.72	8.72
Pumping unit	8.72	--

TABLE 6-16 . Person- and Machine-Day Requirements For Cement Plant Operation (Day/50 m^3)

HE operator	1
S. laborer	2
Laborer	1
QA tech	0.04
Front end loader	1

climate, and (2) maintenance of disposal cells. The personnel-day requirements per 1,000 m^3 of accumulated waste for maintenance of structures and grounds, and radiation surveys, are presented in Table 6-17. Similarly, personnel- and machine-day requirements per 1,000 m^3 of accumulated waste for the maintenance of disposal cells are presented in Table 6-18.

TABLE 6-17 . Facility Maintenance Requirements - 1

Personnel	Repackaged Disposal Option	All Other Options
QA tech	0.15	0.075
Laborer	0.1	0.05
S. laborer	0.2	0.1
Radtech	0.15	0.075

TABLE 6-18 . Facility Maintenance Requirements - 2

Personnel or Machine	Unstable Cells		Stable Cells	
	Humid	Arid	Humid	Arid
QA tech	0.05	0.025	0.00125	0.000625
Laborer	0.1	0.05	0.0025	0.00125
S. laborer	0.05	0.025	0.00125	0.000625
Front end loader	0.1	0.025	0.025	--
Farm tractor	0.1	--	0.1	--

Environmental monitoring personnel requirements are estimated based on the total environmental monitoring costs (see Section 6.2.3). It is estimated that radtech and QA tech requirements are 0.0075 and 0.0025 person-day, respectively, per dollar of total environmental monitoring costs.

6.2.2 Disposal Cell Materials

Disposal cell material cost requirements are considered in Appendix C. These costs are in three groups: basic costs, backfill costs and other material costs. Basic costs for each disposal cell are presented in Table 6-19. These costs are based on the unit material costs given in Table 6-20. Unit material costs may be changed by the code user; however, this will require recalculation of the costs presented in Table 6-19.

TABLE 6-19 . Disposal Cell Materials Unit Costs ($)

ID Index	Disposal Cell	Reference Cover	Improved Cover
1	6,792	1,087	16,185
2	1,272	147	2,195
3	447	4,602	9,203
4	349	624	1,248
5	9,459	648	9,648
6	1,455	2,743	5,486
7	712	79	1,182
8	260	336	672
9	278,755	353	8,203
10	271,974	1,494	2,987
11	161,899	232	5,975
12	154,469	983	1,666
13	182,341	1,134	32,004
14	182,343	349	9,601

Table 6-20. Unit Material Costs

Material	Cost ($)	Unit	Material	Cost ($)	Unit
Sand	4	m^3	Gravel	4	m^3
Crushed Stone	4	m^3	Standpipes	7	m
Access Boxes	140	each	Markers	20	each
Monuments	200	each	Concrete	45	m^3
Rebar	1.3	m	Formwork	10	m^2
Asphalt	6.5	m^2	Sealant	1.3	m^2
Seed	0.18	m^2	Cobbles	5	m^3
Clay	5	m^3	Grout	45	m^3

The second portion of the disposal cell material costs is associated with backfill material costs. Unit material costs ($\$45/m^3$ for grout and $\$5/m^3$ for sand/gravel) are multiplied by the backfill (WB) and the plug backfill (PB) volumes, for concrete slit trenches and concrete trenches. For all other disposal technologies, soil is asssumed to be used for the plug backfill, and these unit material costs are multiplied only by WB.

The third component of the disposal cell material costs covers additional materials used during the repackaging disposal option -- i.e., disposal blocs and grout used to fill the voids within the blocs. The total cost is assumed to be $960 per filled bloc. This results in a cost of about $\$243$ per m^3 of site-processed waste.

6.2.3 Environmental Monitoring

Environmental monitoring and sampling costs are composed of base costs plus additional sampling costs relating to (1) atmospheric monitoring, and (2) monitoring onsite wells and sumps. Base costs are calculated depending on the annual volume of waste and environmental conditions as given in Table 6-21. Atmospheric sampling costs are calculated in accordance with Table 6-22. For onsite wells and sumps, annual costs are calculated as a function of the cumulative waste volume disposed up to that year multiplied by the unit costs given in Table 6-23.

TABLE 6-21 . Base Environmental Monitoring Costs

Site Environment	Annual Waste Volume - m^3		
	2,000	25,000	50,000
Arid	40,000	60,000	80,000
Humid	60,000	80,000	100,000

TABLE 6-22 . Atmospheric Sampling Costs

Basic Cost	42,000
Chemical waste segregation	+14,000
Unstable waste segregation	+14,000
Disposal of Class D waste	+14,000
Extreme stabilization	+70,000

TABLE 6-23 . Unit Sampling Costs for Onsite Wells and Sumps ($\$/m^3$)

Environment	Stable	Unstable
Arid	0.003	0.03
Humid	0.03	0.3

6.2.4 Personnel Training and Monitoring

Personnel training costs depend on the type of work performed by the employee. Badged employees (see Table 6-8) wear radiation detection badges and work in radiation areas, whereas unbadged employees do not routinely enter the operational area. The badged employees are expected to receive a more intensive training and are also assumed to be subjected to a more intensive personnel monitoring program. This cost component is calculated by first determining the numbers of unbadged and badged employees. These numbers are then multiplied by units costs of $500/year and $2,100/year, respectively.

6.2.5 Heavy Equipment Operating Expenses

These costs are dependent on the number of heavy equipment machine-days calculated to be needed (see Section 6.2.1). Operating expenses are calculated based on the unit costs given in Table 6-24.

6.2.6 QA and Compliance Testing

These costs are primarily dependent on the number of disposal cells established during the year. They are calculated based on the unit costs given in Table 6-25.

6.2.7 Other Costs

This cost component includes eleven items: administration (parent company and state or compact), regulatory costs, consulting and studies, legal fees, public outreach, insurance, equipment replacement, miscellaneous expenses, utilities, maintenance, and contingency. The first six of these costs are relatively constant, and are as follows.

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TABLE 6-24 . Equipment Operating Costs ($/machine-day)

Bulldozer	210	Auger Rig	200
Front end loader	140	Paving machine	85
Dump truck	100	Tandem roller	40
Pan scraper	280	Compactor	100
Motor grader	130	Forklift, large	60
Backhoe	45	Farm tractor	25
40-ton crane	115	Hand tamper	3
100-ton crane	250	Cement truck	25
Forklift, small	50	Cement pump and pipes	220
Water truck	70		

TABLE 6-25 . Quality Assurance and Compliance Test Costs (per Disposal Cell)

ID	Cost($)	ID	Cost($)	ID	Cost($)
1	500	6	200	11	2,500
2	200	7	200	12	2,500
3	500	8	100	13	3,000
4	200	9	2,500	14	3,000
5	300	10	2,500		

Administration costs include the local office costs of the parent company, $75,000/year, state or compact administration costs, $100,000/year, and parent company corporate support which is assumed to depend on the annual volume in the manner shown in Table 6-26.

TABLE 6-26 . Corporate Support Costs ($)

Annual Volume (m^3/year)				
2,000	5,000	20,000	35,000	50,000
40,000	100,000	200,000	350,000	500,000

Regulatory costs are assumed to consist of a constant $170,000 per year, plus $200,000 every 5 years (starting with the fifth year of operation) for license renewal, plus $200,000 for preparation of a final closure plan during the final year of operation. Consulting and studies costs are assumed to be $100,000/year. Legal fees are assumed to be $150,000/year. Public outreach is assumed to be $100,000/year. Insurance (nuclear liability and other insurance) is assumed to be $150,000/year.

Of the remaining five components, equipment replacement has already been considered in Table 6-4. All the heavy equipment is replaced according to this schedule. Miscellaneous expenses include costs resulting from communications, travel, temporary equipment rental, moving expenses, etc. Utilities include all expendables. These last two items are given as a function of waste volume in Table 6-27.

TABLE 6-27 . Utilities and Miscellaneous Expenses

Waste Volume - m^3	Utilities	Miscellaneous
<10,000	$30,000	$150,000
10,000 - 20,000	40,000	200,000
20,000 - 30,000	50,000	250,000
30,000 - 40,000	60,000	300,000
40,000 - 50,000	70,000	350,000
>50,000	80,000	400,000

Maintenance includes material costs for routine upkeep of site grounds, buildings, roads, fencing, lighting, etc. It is estimated to equal 1% per year of the capital expenditures for these items. Finally, the contingency item is estimated to equal 20% of all of the above costs.

6.3 Closure Period

In this report only one closure alternative is considered in detail, namely, minimum closure activities. The closure period is therefore assumed to last 1 year.

The disposal cells are assumed to have been "closed" during operations, so that only decontamination and demolition of site structures and buildings is required. Consequently, the major factor contributing to the closure costs is disposing of the radioactive waste generated as a result of the decontamination and demolition activities. This waste volume is assumed to be 3400 m^3 for the repackaged disposal option, and 1130 m^3 for all other options. Moreover, this waste is assumed to be unstable Class A waste and for purposes of handling is assumed be regular care level. In addition, it is assumed to be packaged in 55-gallon drums, and is disposed using the unstable Class A waste disposal technology.

In a manner similar to the operational period, closure cost components have been identified, and are discussed in Appendix C. These costs will be discussed in this section in three groups: salaries, equipment expenses, and other costs.

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6.3.1 Salaries

In a manner similar to the operational period, salaries include those for site administration, support, and workers. Unit salaries are assumed to be identical with those in the operational period. Six badged personnel with a salary plus benefits requirement of $226,200 are assumed for the administrative and support personnel. Worker requirements are calculated in a manner similar to the operational period. Items that contribute to worker requirements are as follows: disposal cell construction, cement plant operation, waste processing, waste emplacement, waste backfilling, disposal cell covering, facility maintenance and surveys, environmental monitoring, and quality assurance. All these items, except waste emplacement, are calculated using the above estimated waste volume and the algorithms outlined in the operational section. For waste emplacement, the personnel- and machine-day requirements given in Table 6-28 are used.

TABLE 6-28 . Waste Emplacement Requirements

Alternative	Person- and Machine-Minutes
Repackaged disposal	41,500
Ref. trench, small humid, and small arid trenches	
Random	16,950
Stacked	67,800
All Other Methods	
Random	22,600
Stacked	73,450

6.3.2 Equipment Expenses

During the closure period, equipment machine-day requirements are determined along with personnel-day requirements. All the equipment is assumed to be leased using the unit costs given in Table 6-29.

6.3.3 Other Costs

These costs include the following fourteen items: nonradiological demolition, administration, regulatory costs, consulting and studies, legal fees, public outreach, disposal cell materials, environmental monitoring, personnel monitoring, miscellaneous expenses, utilities, insurance, QA and compliance testing, and contingency. These are considered below.

Nonradiological demolition is assumed to be contracted, and is assumed to cost a constant $100,000. Administration costs are separated into two components: (1) parent company support (includes local office) is assumed to cost $150,000 for the repackaging alternative, and $100,000 for other alternatives, and (2) state or compact administration is assumed to cost

TABLE 6-29 . Equipment Unit Costs ($/machine-day)

Bulldozer	890	Auger Rig	600
Front end loader	420	Paving machine	355
Dump truck	215	Tandem roller	170
Pan scraper	940	Compactor	300
Motor grader	420	Forklift, large	160
Backhoe	245	Farm tractor	50
40-ton crane	480	Hand tamper	12
100-ton crane	835	Cement truck	50
Forklift, small	140	Cement pump and pipes	460
Water truck	200		

$150,000. Regulatory costs are assumed to be a constant $135,000, consulting and studies $100,000, legal fees $150,000, and public outreach $100,000.

Disposal cell materials depend on the technology used to dispose the waste generated during closure (see Table C-21 of Appendix C). Environmental monitoring costs are assumed to be the same as those for the final year of site operation. Personnel monitoring is assumed to be $600/person applied to all of the closure personnel. Miscellaneous expenses are assumed to be a constant $150,000, utilities are assumed to cost $30,000, and insurance is assumed to cost $150,000. QA and compliance testing is calculated using the assumed radiological demolition volume and Table 6-25. Finally, contingency is taken to be 20% of all of the above components.

6.4 Surveillance Period

During the surveillance period, the licensee would still be responsible for all site maintenance activities until the site license is transferred to the site owner. There appears to be three general variables that influence costs during this period: total waste volume (size of the site), site environment (humid or arid), and stability of the disposed waste.

In a manner similar to the operational period, surveillance period cost components have been identified and are discussed in Appendix C. These costs will be discussed in this section in four groups: salaries, environmental and personnel monitoring, equipment expenses, and other costs. These are discussed below.

6.4.1 Salaries

Personnel contributing to this cost component consists of 1 full-time site employee plus some part time personnel. The full-time employee is assumed to be equivalent to a foreman. Thus, the costs for this component become $36,400/year. Part-time personnel are required for facility maintenance and surveys, and environmental monitoring.

Facility maintenance and survey costs are handled in a manner similar to the operational period requirements (see Section 6.2.1). This cost has two components: the first component is given in Table 6-30.

TABLE 6-30 . Facility Maintenance Requirements - 1
(per 1,000 m^3 of accumulated waste)

QA tech	0.0375
Laborer	0.025
S. laborer	0.05
Radtech	0.0375

The second component is estimated using Table 6-18, except that no heavy equipment is involved. Part-time personnel requirements for environmental monitoring are also handled in a manner similar to operational requirements, i.e., it is estimated that 0.0075 person-day of radtech and 0.0025 person-day of QA tech will be required for each dollar of environmental monitoring costs.

6.4.2 Environmental and Personnel Monitoring

These costs are assumed to be a function of the total volume of waste disposed, the site environment, and the stability of the disposed waste. Table 6-31 is used to determine environmental monitoring sampling costs.

TABLE 6-31 . Environmental Monitoring Costs ($)

Site Environment and Waste Stability	Total Waste Volume Disposed - m^3 50,000	500,000	1,000,000
Arid-unstable	41,500	75,000	110,000
Arid-stable	40,150	61,500	83,000
Humid-unstable	75,000	230,000	400,000
Humid stable	61,500	95,000	130,000

This table gives annual costs assuming that the waste is disposed entirely in either a stable or an unstable manner. For a mixture of the two cases, respective costs are prorated, e.g., assuming VU and VS denote the unstable and stable waste volumes, respectively, of a 500,000 m^3 facility located in an arid environment, environmental monitoring costs are calculated as follows:

$$\text{Cost} = 75{,}000 \times VU/(VU+VS) + 61{,}500 \times VS/(VU+VS) \qquad (6\text{-}7)$$

Personnel monitoring costs are calculated based on the total number of personnel that have spent some time at the site. A unit cost of $600/year per person-year is applied.

6.4.3 Equipment Expenses

This cost component is handled in a manner similar to the closure costs. The vehicles are assumed to be leased using the unit costs given in Table 6-32. Maintenance costs are included.

TABLE 6-32 . Unit Equipment Costs ($/machine-day)

Pickup truck	70
Front end loader	420
Farm tractor	50

Machine-day requirements are as follows: one full time pickup truck is required, i.e., 250 machine-days which implies an annual cost of $17,500. For the other equipment Table 6-33 is used.

TABLE 6-33 . Machine-day Requirements (machine-days/1000 m^3)

Front end loader	Humid-unstable:	0.25
	Humid-stable :	0.06
	Arid-unstable :	0.06
	Arid-stable :	0.0
Farm tractor	Humid-unstable:	0.25
	Humid-stable :	0.25
	Arid-unstable :	0.0
	Arid-stable :	0.0

6.4.4 Other Costs

These costs consists of two items: constant costs and contingency. Constant costs are the sum of administrative costs, regulatory costs, and other costs (e.g., legal fees, insurance, utilities) as detailed in Appendix C. The total costs are estimated at $520,000 per year. Finally, contingency is assumed to be 20% of all of the above cost components.

6.5 Institutional Control Period

During this period, the site is assumed to be under the control of a state agency. There are two major cost components in this period. Variable costs and other costs. Variable costs include salaries, environmental monitoring, personnel monitoring, and equipment costs. Variable costs are assumed to be the same as those for the last year of the surveillance period. Other costs are composed of constant costs (administration, regulatory, and other costs totaling $330,000/year) and contingency which is assumed to be 20% of the sum of the variable and constant costs.

APPENDIX A

WASTE DESCRIPTIONS, VOLUMES AND ACTIVITIES

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APPENDIX A. Waste Descriptions, Volumes, and Activities

This appendix, in conjunction with Appendix B, presents a description of the radioactive waste source term considered in this report. This radioactive waste source is an update of that developed by Oztunali and Wild *et al.* (Refs. 1, 2) and used by NRC in the draft and final environmental impact statements (EIS) (Refs. 3, 4) for the regulation 10 CFR Part 61 ("Licensing Requirements for Land Disposal of Radioactive Waste"). This radioactive waste source term consists of a great variety of different types and forms of waste, and includes waste which ranges in activity from low to very high levels. In this report the higher activity wastes are emphasized--i.e., those wastes which have a reasonable potential for containing radionuclides in concentrations which are close to or exceed Class C limits as defined by 10 CFR Part 61 (Ref. 5).

This appendix is organized as follows. The first chapter presents an overview of the development of the source term, including major assumptions, its structure, and its intended use. This chapter also includes an overview of radioactive waste generators and the abbreviations used to signify all the waste streams considered in this report.

Volume and gross activity projections for each waste stream are then presented in following Chapters A.2 through A.8. These projections are typically for "untreated" waste, which means waste in an as-generated form prior to further treatment such as solidification or incineration, and prior to packaging. This is followed by Chapter A.9, which presents projected radionuclide concentrations for each of the "untreated" waste streams.

Closely allied with Appendix A is the following Appendix B. Appendix B addresses the various waste processing options that will be considered in the report, and also describes how the waste volumes, gross activities, and radionuclide distributions change based on application of the different waste processing options. Appendix B also addresses the concept of "waste spectra," which denotes the total volumes and properties of the waste streams following processing by a set of selected waste treatment options. Each spectrum corresponds to a general level of waste performance.

A.1 INTRODUCTION

This chapter is divided into three sections. Section A.1.1 presents overall concepts, Section A.1.2 presents an overview of waste generators, and Section A.1.3 presents the notation used to identify each waste stream.

A.1.1 Concepts

What has been termed low-level radioactive waste, or LLW*, is generated by more than 20,000 NRC and Agreement State licensees throughout the country. Such licensees include operators of nuclear fuel cycle facilities such as nuclear power reactors, reactor fuel fabrication plants, and uranium hexafluoride (UF_6) conversion plants. Waste is also generated by non-fuel cycle licensees such as hospitals, medical research institutions, colleges and universities, industrial research laboratories, facilities involved in the production of radiopharmaceuticals, and other industrial uses of radioactive materials. The waste thus produced by both fuel cycle and non-fuel cycle waste generators is very diverse in terms of volume, activity, and other physical, radiological, and chemical characteristics. It essentially includes everything that is discarded as waste and ranges from hospital trash that is only suspected of being contaminated to highly radioactive activated structural components from nuclear power reactors. These physical, radiological, and chemical characteristics may furthermore be altered by the waste generator as the waste is processed and/or packaged into a form which is suitable for shipment to an LLW disposal facility.

This great heterogeneity in waste forms and characteristics presents a number of difficulties in modeling LLW processing and disposal. A regulator, for example, may be interested in performing an anlysis of the costs and impacts of alternative requirements for LLW disposal. Depending upon the form and other characteristics of the waste, and upon the available technologies for waste processing and packaging, each alternative requirement may result in varying levels of environmental impacts. Each alternative requirement may also result in varying levels of economic and radiological costs to individual licensees. Despite this, previous studies on LLW management and disposal have frequently assumed LLW to be a mostly uncharacterized homogeneous mass with little attempt to distinguish, in a quantitative manner, the different waste types and forms. This has resulted in erroneous conclusions being formulated regarding the costs and other impacts of waste disposal, and of preferred regulatory requirements.

The approach adopted in this report is similar to that taken earlier in references 1 and 2--i.e., the heterogeneous nature of LLW is considered directly, and in a form suitable for analysis. Various types of waste are aggregated together into groups having common sources of generation and physical, radiological, and chemical characteristics. These groups are termed "waste streams" in this report, and an example waste stream consists of cartridge filters from pressurized water reactors; another example waste stream consists of biological waste generated by hospitals and other institutions. Thirty-six waste streams were considered in the draft Part 61 EIS (Ref. 3), while 37 waste streams were considered in the final Part 61 EIS (Ref. 4). This report considers 148 waste streams.

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*For the purposes of this appendix, LLW has the same meaning as that given in 10 CFR Part 61 and the Low-Level Radioactive Waste Policy Act: radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in Section 11e.(2) of the Atomic Energy Act (uranium or thorium tailings and waste).

For each waste stream, projections are made of the volumes annually estimated to be generated from 1981 to the year 2030, as well as the average gross radionuclide activity anticipated in each stream. (For some waste streams, the gross activity is given as a distribution over the waste stream volume, rather than an average.) Each waste stream is also characterized by a particular radionuclide distribution over the gross activity--i.e., for each waste stream the radionuclides are assumed to contained in the waste stream in set proportions to one another. (The proportions vary depending upon the waste stream.)

The waste volume projections are furthermore established on a regional basis. To establish the regional projections, the contiguous United States is assumed to be divided into four regions as shown in Figure A.1. The four regions considered correspond to the five U.S. Nuclear Regulatory Commission regions and are termed the northeast region (NRC Region I), the southeast region (NRC Region II), the midwest region (NRC Region III), and the western region (NRC Regions IV and V). Each region could represent a large multistate compact formed for waste disposal--e.g., the northeast region consists of 11 states, the southeast region 10 states, the midwest region 8 states, and the western region 19 states. (The larger size of the western region is principally due to the lower volumes of waste projected to be generated in NRC Regions IV and V.)

At the time of the development of the data base reports (Refs. 1, 2), the state compacting state process initiated by the Low-Level Radioactive Waste Policy Act of 1980 was just being started. At that time it seemed appropriate to consider very large regions since this would be much more cost-effective than smaller regions. Disposal charges are inversely related to disposal volume--the larger the waste volume to be disposed, the lower the disposal cost. More recent indications (e.g., Ref. 6) are that there may be a number of small compacts, some consisting of only one or a few states. This implies that a better approach would be to address waste projections on a state basis. This was not done at this particular time since the resulting complexity in updating the Part 61 analysis methodology would go far beyond the resources currently available. It would be a very good subject for follow-on work.

Additional important concepts include (1) the changes in waste volumes and concentrations according to different waste processing and packaging options, (2) the manner in which gross physical and chemical characteristics are considered in the analysis, and (3) the use of different waste spectra.

The function of this analysis methodology is not only to compare the radiological impacts of waste disposal under different alternatives, but also to compare the costs and other impacts of alternative waste packaging and disposal schemes. For this reason, the projections of waste are frequently given as volumes in an as-generated form prior to further processing (e.g., solidification) and packaging for shipment. These are somewhat inaccurately termed "untreated" volumes. For example, volumes for untreated trash waste streams are given as-generated prior to such possible processing options as compaction or incineration. Concentrated liquid wastes from nuclear power plants are given as evaporator bottoms prior to solidification. This approach is not fully used for many waste streams, however, based on a lack of specific data on the individual waste processing and packaging habits of each of 20,000 licensees.

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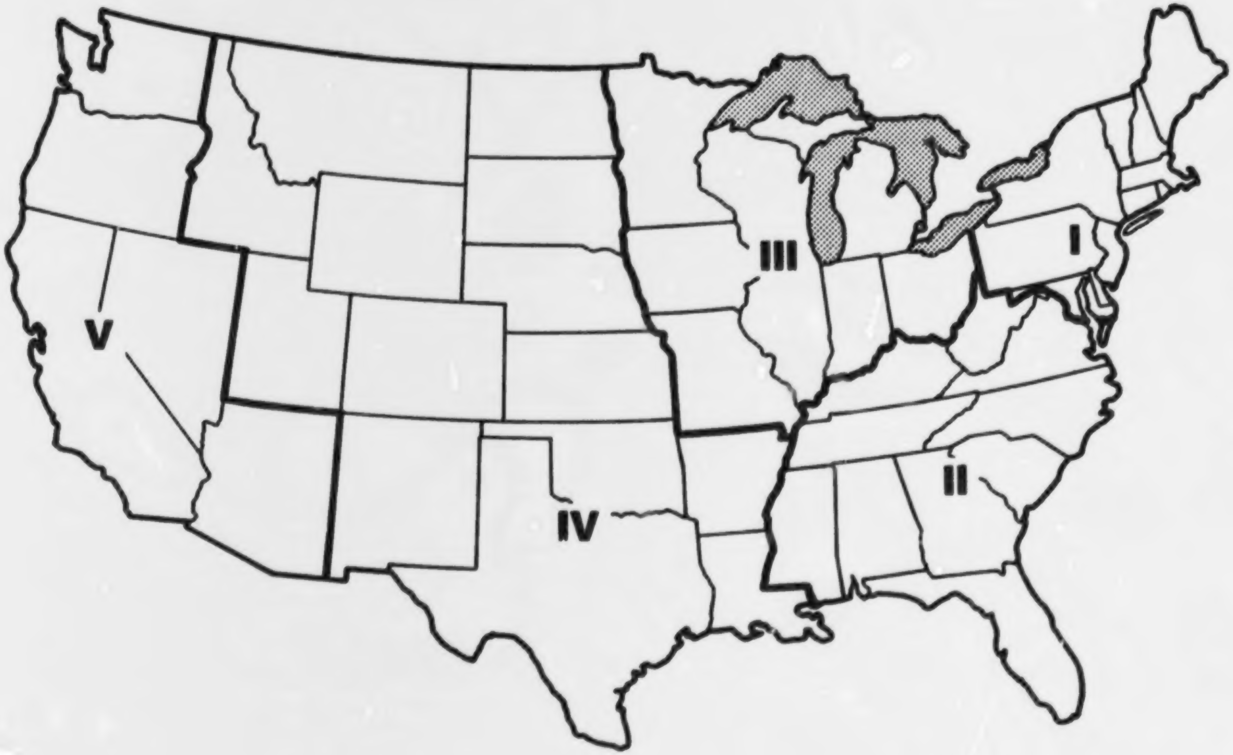

Figure A.1. Low-Level Waste Generation Regions

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Each waste stream may be subjected to a number of different waste processing and packaging options, and for each option the final as-shipped waste volume is determined, as well as the gross physical and chemical characteristics. The final as-shipped waste volumes are determined by multiplying the untreated waste volumes by a volume increase factor (VIF), and then dividing the result by a volume reduction factor (VRF). For example, assume that the untreated volume of a particular trash waste stream is 100,000 m^3, and that this waste stream is incinerated and the resulting ashes solidified in cement in 55-gallon drums. Assuming that incineration reduces the trash volume by a factor of 20, and the solidification process increases the ash volume by a factor of 1.4, the final as-shipped volume of this particular waste stream under this particular processing and packaging option is $100,000 * (1.4/20) = 7,000$ m^3.

For each waste processing and packaging option, the radionuclide concentrations within the waste stream are also determined using the VIF and VRF factors.

The gross physical and chemical characteristics of the waste stream are considered for each processing and packaging option using a set of waste form behavior indices. The indices are listed in Table A-1 and are discussed in more detail in Appendix B. Briefly, however, the integers are used as a qualitative ranking of a particular property of the waste, and also to trigger specific computational procedures in the impacts analysis computer codes. These waste form behavior indices are used in conjunction with a set of waste processing indices to determine costs of impacts of waste processing, transportation, and disposal.

Many of the waste form behavior indices rank particular waste forms in terms of tendencies to behave in a certain way. For example, the flammability component of the accident index compares different waste forms in terms of their tendency to burn, the dispersibility index compares different waste forms in terms of their tendency to disperse into the air, and so forth. Many of the integer values assigned to these rankings are done so in a rather subjective way. However, this is not believed to be a major problem since the major purpose of the impact analysis methodology is to enable the user to compare, and quantify to a certain extent, trends associated with particular alternative rulemaking actions. The waste form behavior indices should not be used as a substitute for real data and test results for specific waste streams.

It is apparent that if a large number of individual waste streams are considered, and each waste stream may have a number of alternative processing and packaging options, then one is faced with an unmanageable number of alternatives which could be considered. To limit the number of these alternatives, the waste processing and packaging options are grouped into five generic options, called waste spectra. Each waste spectra represents a relative level of processing and packaging applied to all the waste streams considered.

Briefly, waste spectrum 1 characterizes past and, in some cases, existing waste management practices. It characterizes a condition in which compressible waste streams are subjected to compaction, but high activity waste streams are disposed for the most part in an unstable waste form. This waste spectrum corresponds most closely to disposal conditions prior to promulgation of the Part 61 regulation. Waste spectrum 2 characterizes improvements in the form of the waste through processing and reduction in waste volume with relatively modest expenditures of time and money. Waste spectrum 3 characterizes further waste form

Table A-1. Waste Form Behavior Indices

Parameter	Symbol	Indices
Accident/Scatter	(I4/ISC)	0 = severe 1 = moderate 2 = slight to moderate 3 = low
Accident/ Flammability	(I4/IFL)	0 = flammable (supports burning) 1 = burns if heat supplied (does not support burning) 2 = low flammability (mixture of material with indices of 0 and 2) 3 = nonflammable
Dispersibility	(I5)	0 = severe 1 = moderate 2 = slight to moderate 3 = low
Leachability	(I6)	1 = unsolidified waste form 2 = solidification scenario A 3 = solidification scenario B 4 = solidification scenario C
Chemical Content	(I7)	0 = no chelating agents or organic chemicals 1 = chelating agents or organic chemicals are likely to be present in the waste form
Stability	(I8)	0 = structurally unstable waste form 1 = solidified waste form 2 = structurally stable solidified waste form 3 = stabilized by placement into HIC 4 = stabilized by another means
Activated Metal	(I9)	0 = not activated metal waste 0 < activated metal waste
Sources	(I10)	0 = not source waste 0 < source waste

improvements and volume reduction at further increased costs, including incineration of most combustible waste streams. Waste spectrum 4 characterizes the maximum volume reduction of improved waste forms that can currently be practically achieved. Waste spectrum 5 characterizes (for most waste streams) the use of containers providing structural support to achieve waste form stability rather than processing to a solid form. The five waste spectra can be used singly or in combinations to represent a particular alternative requirement.

A.1.2 Overview of Waste Generators

Nuclear Fuel Cycle Sources

Nuclear fuel cycle waste generators include facilities involved in the commercial generation of electrical power through the use of nuclear energy. The current fuel cycle is based upon once-through use of uranium fuel as conceptually depicted in Figure A.2 (Ref. 7).

The nuclear fuel cycle begins with mining and milling of uranium ore. Uranium ore is generally obtained from either open pit or underground mines and is usually transported to a centralized mill for processing. Uranium mills convert uranium ore to yellowcake, which primarily consists of U_3O_8. Disposal of liquid and solid wastes generated as part of milling operations is addressed elsewhere and is not considered further in this report.

Yellowcake produced from milling operations is then shipped to conversion plants that convert U_3O_8 to uranium hexafluoride (UF_6). The conversion process generates liquid and solid waste streams, most of which are recycled to recover uranium prior to storage in onsite ponds or reuse within the plant. Onsite storage at conversion facilities is presently regulated by NRC under 10 CFR Part 40. Small quantities of low-activity wastes contaminated with natural uranium are shipped offsite to licensed near-surface disposal facilities. These wastes are considered further in this report. Currently, there are two UF_6 conversion plants in operation in the United States; one plant is located in NRC Region III and one in NRC Region IV.

Following conversion, natural UF_6 is shipped to enrichment facilities for enrichment in fissile U-235. In this process, the U-235 content of the uranium is raised from natural concentrations (about 0.7 weight percent) to about 2 to 4 weight percent. Currently, three enrichment plants using the gaseous diffusion process are in operation and these are located at Portsmouth, Ohio; Paducah, Kentucky; and Oak Ridge, Tennessee. These plants are owned and operated by the federal government and wastes produced from plant operation are not sent to commercial disposal facilities. Hence, waste streams produced from uranium enrichment operations are not considered further in this report.

Enriched UF_6 is then shipped to commercial fuel fabrication plants which convert the enriched UF_6 to uranium dioxide (UO_2) powder, produce UO_2 pellets, fabricate fuel rods containing the UO_2 pellets, and combine the fuel rods into fuel assemblies for use in light water reactors (LWR). Most of the liquids, sludges, and other wastes produced during the UF_6-to-UO_2 conversion process are presently being stored at the fabrication plants, although some wastes in the form of dry solids (principally CaF_2) contaminated with low levels of enriched uranium are being shipped offsite for disposal. Low-activity waste, principally trash, is also generated during the pelletizing and subsequent

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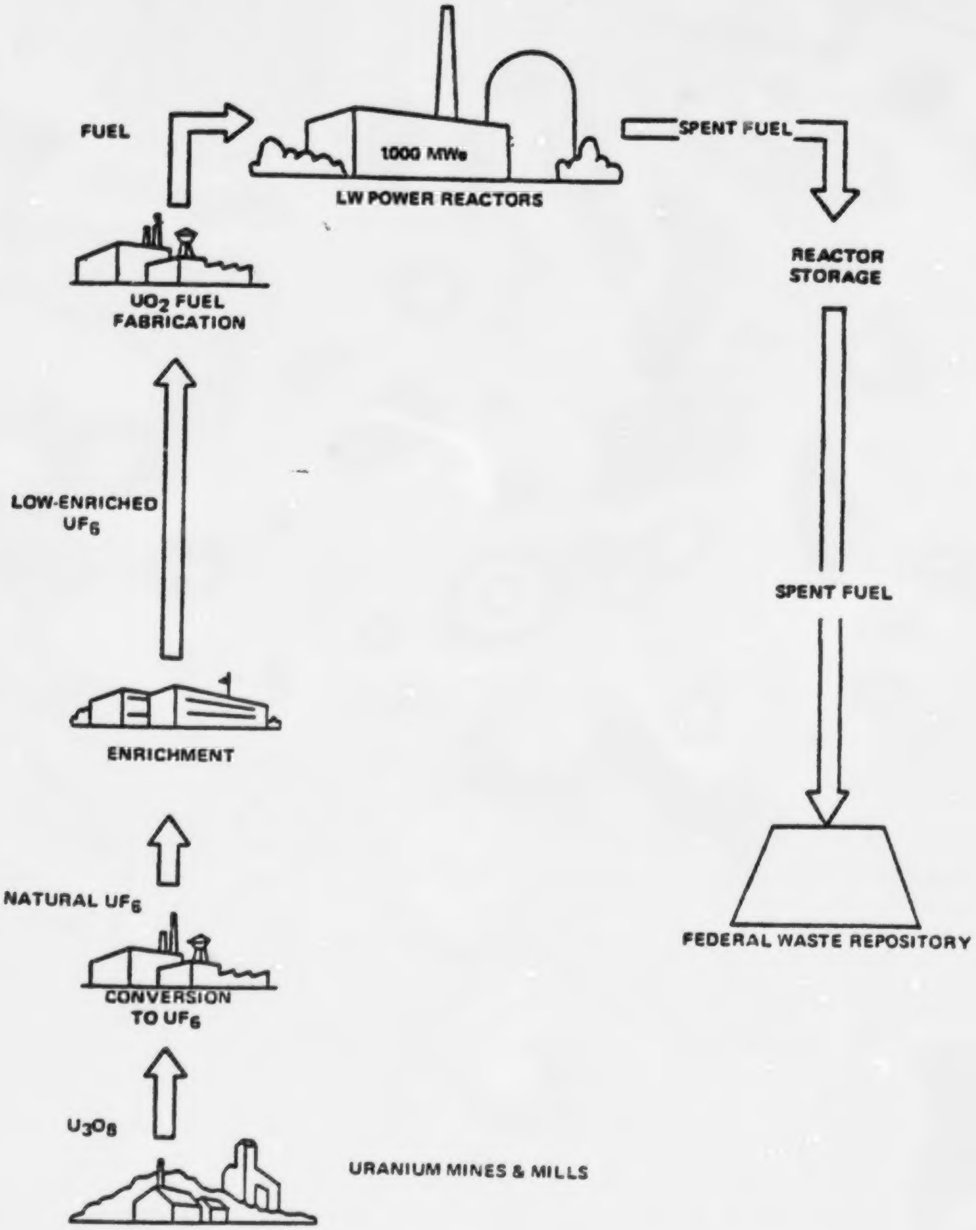
FUEL
1000 MWe
LW POWER REACTORS
SPENT FUEL
REACTOR STORAGE
UO_2 FUEL FABRICATION
LOW-ENRICHED UF_6
SPENT FUEL
ENRICHMENT
FEDERAL WASTE REPOSITORY
NATURAL UF_6
CONVERSION TO UF_6
U_3O_8
URANIUM MINES & MILLS

Figure A.2. Once-Through Uranium Fuel Cycle

fabrication processes, and these waste streams are also shipped offsite for disposal.

Fuel assemblies are then shipped to central station nuclear power plants, almost all of which utilize light water power reactors for production of electrical power through use of the energy released during fission of the uranium fuel. Approximately two out of every three power plants use pressurized water reactors (PWRs) for energy generation while the remainder use boiling water reactors (BWRs). During operations, waste is generated in a number of forms having specific activities ranging from low to moderately high levels. Much, if not most, of the waste is generated as a result of operating and maintaining plant processes which maintain concentrations of radiocontaminants in the reactor coolant and other process systems to low levels and reduce effluent releases from the plant to acceptable levels. The presence of such radiocontaminants in reactor cooling systems can result from activation of corrosion products or from leakage of fission products out of the fuel rods. The treatment and maintenance operations result in wet wastes such as filter sludges, spent resins, and evaporator bottoms, as well as compactible and noncompactible dry wastes. Liquids such as evaporation bottoms are solidified, while other wet wastes such as ion-exchange resins are generally dewatered and packaged into containers for shipment. Some compaction is usually performed on compactible trash. The wastes are then generally shipped offsite for disposal.

The fuel used in the reactors must be periodically replaced. Generally about one-third of the fuel in the reactor core is replaced approximately every 12 to 18 months. Most of this spent fuel is stored at the power stations within large spent fuel holding pools. Such storage operations generate similar types of waste as that discussed above. A small fraction of this fuel, however, has been stored offsite in fuel pools located within two facilities originally designed to reprocess the fuel. One facility (the Nuclear Fuel Service plant in West Valley, NY) suspended reprocessing operations in 1971, and the spent fuel stored there is currently being shipped back to the parties which generated the spent fuel. This is being done as an initial step in the West Valley Demonstration Project, which is intended to decontaminate the facility and solidify stored liquid high-level waste. (See below.) The other facility (the General Electric facility in Morris, IL) never became operational. Additional facilities specifically constructed for storage of spent fuel may be constructed in the future; these may be located either at the operating reactors' sites or at away-from-reactor sites.

Some of the nonfuel metal components within the reactor vessels must also be occasionally replaced. These include poison curtains, control rods, flux wires, and other hardware. These activated metal wastes are frequently of high specific activities.

Two other sources of waste can also be more or less attributed to fuel cycle activities, although there are considerable uncertainties in the generation of such waste. The first consists of waste generated from possible future full scale decontamination of light water reactor primary coolant systems. The second consists of waste generated due to decontamination of, and in some cases continued use of, facilities performing experimental studies on light water (and other) reactor fuel. These studies include fabrication of mixed oxide fuel test assemblies as well as fuel burnup studies.

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Non-Fuel Cycle Sources

Non-fuel cycle waste generators include approximately 20,000 facilities licensed by NRC or Agreement State agencies to use radioactive materials. Most non-fuel cycle waste generators may be classified as either institutional or industrial.

Institutional waste generators include hospitals, medical schools and research facilities, colleges, and universities. Waste generation rates and waste characteristics vary significantly between institutional waste generators and it is therefore difficult to consider each type of institution as a separate waste generator. Therefore, all institutional facilities are considered as a single waste source in this report.

Industrial waste generators are also frequently considered as a single source of waste, and include industries which produce and distribute radionuclides, manufacture materials containing radioisotopes for industrial uses, and use radioisotopes in laboratory studies, instruments, devices, and manufacturing processes. Industrial waste generators have not been surveyed to as great an extent as other types of waste generators. The waste generated by industrial generators is also very difficult to characterize. One reason is that a given licensee may generate waste on a very infrequent or one-time basis. Some of this waste can have quite high specific activities. Another reason is that the radionuclides contained in the waste depend upon the particular isotopes used in the manufacturing process, which is highly facility-specific.

Two other non-fuel cycle groups of waste include "discrete" radium sources, as well as waste from military activities. Discrete radium sources include sealed radium sources of various sizes plus radium-contaminated ion-exchange resins generated as a result of operation of groundwater radium removal systems. Government military activities are not licensed by either NRC or Agreement states, although radioactive wastes are occasionally generated which are shipped to commercial disposal facilities for disposal. The most significant fraction of these wastes appears to originate from maintenance of nuclear powered naval vessels.

Non-Routine Sources

In this report, non-routine sources are considered to be those from uranium fuel recycle activities, LWR decommissioning, the West Valley Demonstration Project, and decontamination of the Three Mile Island nuclear power plant, Unit 2. These are termed non-routine sources because, compared with the other waste sources considered earlier, waste from these sources is not considered to be routinely generated over the next 20 years, and/or there are large uncertainties in the volumes and activities in the waste to be generated.

Two basic options are available for the disposition of spent nuclear power plant fuel. One option is to treat the spent fuel as high-level waste and dispose of the spent fuel in a federal repository to be constructed and operated by the Department of Energy (DOE). This is the once-through cycle depicted in Figure A.2. Another option is to recycle the spent fuel as conceptually illustrated in Figure A.3 (Ref. 7).

In this option, spent fuel would be shipped to a reprocessing plant which, using chemical separation processes, would recover residual uranium and plutonium

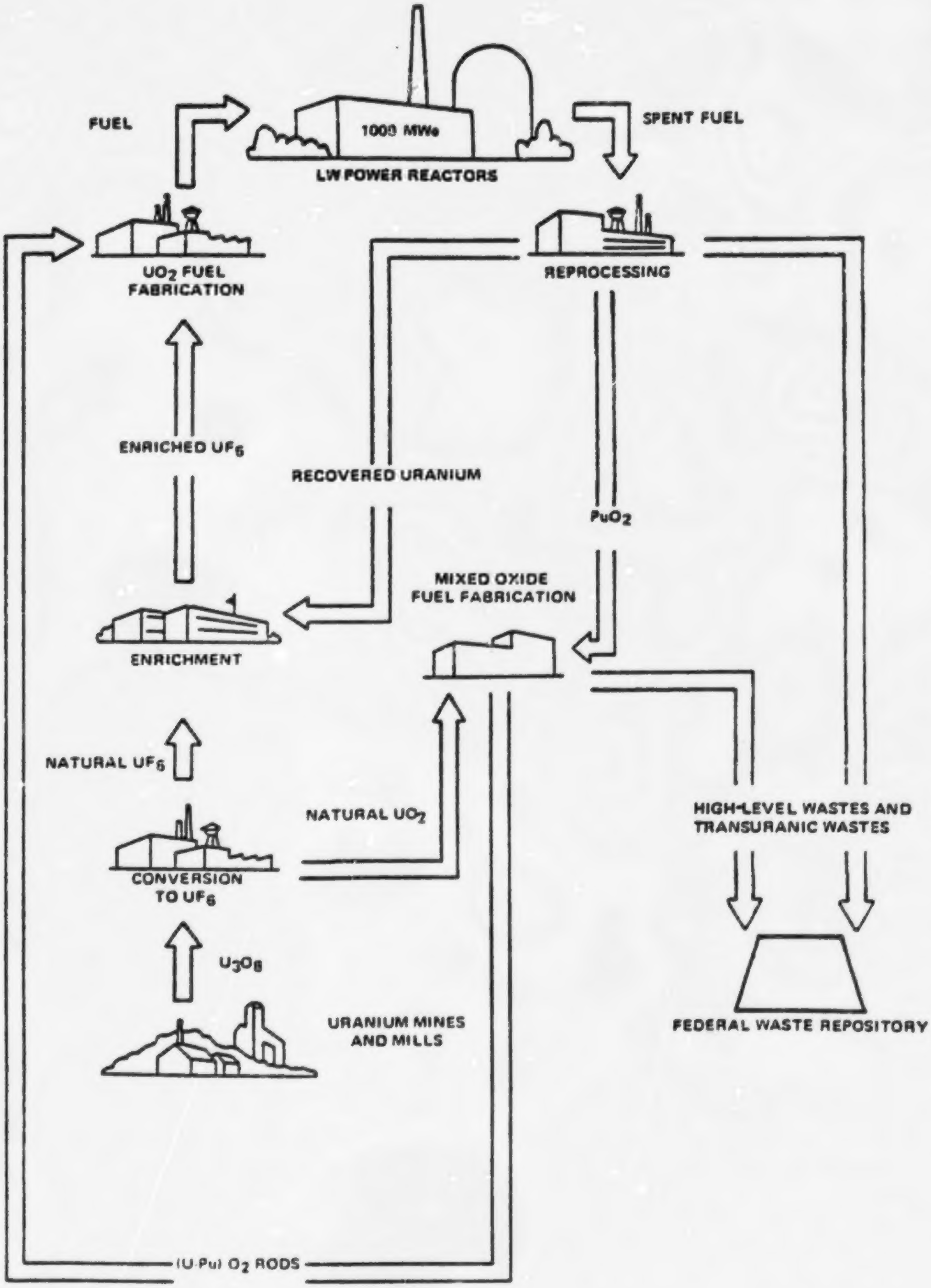
FUEL
1000 MWe
LW POWER REACTORS
SPENT FUEL
UO_2 FUEL FABRICATION
REPROCESSING
ENRICHED UF_6
RECOVERED URANIUM
PuO_2
MIXED OXIDE FUEL FABRICATION
ENRICHMENT
NATURAL UF_6
NATURAL UO_2
CONVERSION TO UF_6
HIGH-LEVEL WASTES AND TRANSURANIC WASTES
U_3O_8
URANIUM MINES AND MILLS
FEDERAL WASTE REPOSITORY
(U-Pu) O_2 RODS

Figure A.3. Uranium and Plutonium Recycle Fuel Cycle

for reuse in reactors. Recovered uranium would be shipped as UF_6 to an enrichment plant for enrichment in U-235. Recovered plutonium would be shipped as plutonium dioxide (PuO_2) powder to a mixed oxide (MOX) fuel fabrication plant where it would be combined with natural UO_2 and fabricated into MOX fuel rods. The MOX fuel rods would then be shipped to a fabrication plant where the MOX fuel rods would be combined with natural uranium fuel rods and assembled into fuel assemblies for reinstallation into LWRs. High-level and transuranic wastes generated during reprocessing and MOX fuel fabrication operations would be shipped to a federal repository for disposal.

For the last several years, the policy of the United States has been to defer the uranium recycle option. There are no reprocessing or MOX fuel fabrication facilities operating in the country and spent fuel removed from nuclear power reactors is currently being stored pending operation of a federal storage facility or repository. It is possible that the country's policy on uranium fuel recycling may eventually change. However, at present, the timing and extent of future fuel reprocessing and MOX fuel fabrication operations are speculative, as is the quantity of waste to be generated through such operations. It is not believed likely by the authors of this report that uranium recycle will be implemented within any reasonable time period--i.e., until after the year 2000 if at all.

A large source of waste to be generated in the future will be from decommissioning light water power reactors. The volumes and activities which will be produced are uncertain, and depend upon such factors as the service life of the plant prior to decommissioning, the size and design of a plant, and the decommissioning mode undertaken (e.g., immediate dismantlement after shutdown vs. deferring dismantlement for up to several years following shutdown). Most of the waste thus produced will be slightly contaminated rubble, with quantities of other waste similar in composition to waste generated during normal operations--e.g., trash, spent resins, filter cartridges, and evaporator bottoms. Some waste, however, mainly consisting of some highly activated metals, are expected to be quite high in radionuclide content.

The only commercial fuel reprocessing facility ever to operate in the United States was located at the Western New York Nuclear Service Center operated by Nuclear Fuel Services (NFS) near West Valley, New York. In 1972, this facility was shut down by NFS and has not operated since. The eventual disposition of the facility, which includes a fuel reprocessing plant, 600,000 gallons of liquid high-level waste stored in a tank, and a waste disposal area, is being addressed at this time through the ongoing West Valley Demonstration Project. This project was mandated by Congress in 1980 by the West Valley Demonstration Project Act, which charges the Department of Energy (DOE) with the responsibility to develop, construct, and operate a high-level liquid waste solidification project at the West Valley plant. This project will solidify the 600,000 gallons of liquid high-level waste presently stored in underground tanks to a final form acceptable for disposal into a Federal repository. Decontamination of existing facilities to prepare for the project as well as activities during the waste solidification project and final decontamination of facilities at the end of the project will generate substantial volumes of waste. Much of this waste is expected to be contaminated with transuranic radionuclides.

The March 28, 1979 accident at the Three Mile Island (TMI) Unit 2 nuclear power station has resulted in considerable damage to the reactor core as well as generation of significant quantities of contaminated water. Removal of damaged core components and other plant equipment, processing of the contaminated water, and decontamination of contaminated plant equipment and surfaces is projected to take several years. Over this time, radioactive wastes in various solid forms will be generated.

Other Waste

This source of waste consists of spent nuclear power fuel. It is apparent that spent nuclear power fuel is not to be disposed as low-level waste; federal regulations prohibit it (Refs. 5, 8). Rather, this material is included in this report's data base to enable a hypothetical comparison of impacts from waste disposal by near-surface or other methods.

Two waste streams are considered. One consists of the fuel rods containing the fission products. Another waste stream includes end pieces, spacers, and other hardware removed from spent fuel assemblies assuming consolidation of the spent fuel rods prior to disposal in a geologic repository.

A.1.3 Waste Streams and Notation

The basic waste streams considered in this report are listed in Table A-2. The waste streams may be consolidated into seven major groups:

- Nuclear power plants
    - Pressurized water reactors
    - Boiling water reactors

- Other nuclear fuel cycle facilities
    - Fuel fabrication plants
    - Uranium hexafluoride conversion plants
    - Plutonium facility decontamination and fuel burnup studies

- Institutional waste

- Industrial waste
    - Large industrial radioisotope manufacturers
    - Large tritium and C-14 manufacturers
    - Small tritium manufacturers and users
    - Sealed sources and devices

- Other non-nuclear fuel cycle waste
    - Radium sealed sources and ion-exchange resins
    - Government and military facilities

- Non-routine waste
    - Uranium fuel reprocessing
    - Mixed oxide fuel fabrication
    - Nuclear power plant decommissioning
    - West Valley Demonstration Project
    - Three Mile Island decontamination

- Other waste
    - Spent nuclear power plant fuel

Table A-2. Waste Groups and Streams

Waste Stream	Symbol
I. Nuclear Power Plants	
PWR ion-exchange resins	P-IXRESIN
PWR concentrated liquids	P-CONCLIQ
PWR filter sludges	P-FSLUDGE
PWR cartridge filters	P-FCARTRG
BWR ion-exchange resins	B-IXRESIN
BWR concentrated liquids	B-CONCLIQ
BWR filter sludges	B-FSLUDGE
PWR combustible trash	P-COTRASH
PWR noncombustible trash	P-NCTRASH
BWR combustible trash	B-COTRASH
BWR noncombustible trash	B-NCTRASH
LWR nonfuel reactor core components	L-NFRCOMP
LWR decontamination waste	L-DECONRS
II. Other Nuclear Fuel Cycle Facilities	
Fuel fabrication process waste	F-PROCESS
Fuel fabrication combustible trash	F-COTRASH
Fuel fabrication noncombustible trash	F-NCTRASH
UF_6 conversion process waste	U-PROCESS
Pu facility decontamination waste	L-PUDECON
Waste from fuel burnup studies in hot cells	L-BURNUPS
III. Institutional Waste	
Combustible trash (large facilities)	I-COTRASH
Combustible trash (small facilities)	I+COTRASH
Absorbed liquids (large facilities)	I-ABSLIQD
Absorbed liquids (small facilities)	I+ABSLIQD
Liquid scintillation waste (large facilities)	I-LIQSCVL
Liquid scintillation waste (small facilities)	I+LIQSCVL
Biological waste (large facilities)	I-BIOWAST
Biological waste (small facilities)	I+BIOWAST
IV. Industrial Waste	
Source and SNM trash (large facilities)	N-SSTRASH
Source and SNM trash (small facilities)	N+SSTRASH
Source and SNM waste	N-SSWASTE
Low activity trash (large facilities)	N-LOTRASH
Low activity trash (small facilities)	N+LOTRASH
Low activity waste	N-LOWASTE
Large industrial radioisotope manufacturers	
High activity waste from isotope production facilities	N-ISOPROD
Low activity trash from isotope production facilities	N-ISOTRSH

Table A-2. Waste Groups and Streams (continued)

Waste Stream	Symbol
Large sealed source manufacturers	N-SORMFG1
	N-SORMFG2
	N-SORMFG3
	N-SORMFG4
Large Tritium and Carbon-14 Manufacturers	
Compactible trash	N-NECOTRA
Absorbed organic liquid	N-NEABLIQ
Solidified aqueous liquid	N-NESOLIQ
Reject product vials	N-NEVIALS
Noncompactible glass	N-NENCGLS
Noncompactible wood/metal	N-NEWOTAL
Tritium gas	N-NETRGAS
Absorbed tritiated liquid	N-NETRILI
Absorbed C-14 liquid	N-NECARLI
Laboratory trash	N-MWTRASH
Absorbed organic liquid	N-MNABLIQ
Solidified aqueous liquid	N-MWSOLIQ
Miscellaneous waste	N-MWWASTE
Small Tritium Manufacturers	
Tritium in paint or as plating	N-TRIPLAT
Gaseous tritium	N-TRITGAS
High-tritium activity scintillation liquid	N-TRISCNT
Tritium in aqueous liquid	N-TRILIQD
Miscellaneous trash	N-TRITRSH
Tritium contained or absorbed in metal	N-TRIFOIL
High Activity Waste	N-HIGHACT
Sealed Sources and Devices	
Tritium sources	N-TRITSOR
Carbon-14 sources	N-CARBSOR
Cobalt-60 sources	N-COBLSOR
Nickel-63 sources	N-NICKSOR
Strontium-90 sources	N-STROSOR
Cesium-137 sources	N-CESISOR
Plutonium-238 sources	N-PLU8SOR
Plutonium-239 sources	N-PLU9SOR
Americium-241 sources	N-AMERSOR
Pu-238 neutron sources	N-PUBESOR
Am-241 neutron sources	N-AMBESOR
V. Other Non-Fuel Cycle Waste	
Radium Sources	
Medical needles	N-RANEEDS
Medical cells	N-RACELLS
Medical plaques	N-RAPLAQU

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Table A-2. Waste Groups and Streams (continued)

Waste Stream	Symbol
Radium Sources (continued)	
Medical nasopharyngeal applicators	N-RANPAPP
Radium-beryllium neutron sources	N-RABESOR
Miscellaneous non-medical sources	N-RAMISCL
Radium ion-exchange resins	N-RARESIN
Navy wet wastes	M-NAVYWET
Navy dry wastes	M-NAVYDRY
VI. Non-Routine Waste	
Uranium Fuel Processing	
High-level liquid waste	R-HLLWFRP
Fuel assembly hardware	R-FUEHARD
Hulls from chop/leach process	R-HULLFRP
Intermediate-level liquid waste	R-ILLWFRP
Silca gel	R-SILIGEL
Main plant high activity compressible trash	R-MPCOTRH
Main plant low activity compressible trash	R-MPCOTRL
Main plant noncompressible trash	R-MPNCTRA
Degraded extractant	R-DEGREXT
Main plant ion-exchange resins	R-MPRESIN
Storage basin resins and filter sludge	R-SBRESIN
Storage basin concentrated liquids	R-SBCOLIQ
Storage basin compressible trash	R-SBCOTRA
Storage basin noncompressible trash	R-SBNCTRA
UF_6 conv. flourinator residues	R-UFFINES
UF_6 conv. K_2UO_4 mud	R-UFK2MUD
UF_6 conv. compressible trash	R-UFCOTRA
UF_6 conv. noncompressible trash	R-UFNCTRA
PuO_2 conv. compressible trash	R-PUCOTRA
PuO_2 conv. noncompressible trash	R-PUNCTRA
Mixed Oxide Fuel Fabrication	
Compressible trash	R-MOXCOTR
Noncompressible trash	R-MOXNCTR
Process plus scrap recovery solutions	R-MOXSOLN
Nuclear Power Plant Decommissioning	
PWR activated core shroud	P-DECORES
PWR activated reactor internals	P-DEACINT
PWR activated reactor vessel	P-DEACVES
PWR activated concrete	P-DEACTCO
PWR contaminated metal	P-DECONME

Table A-2. Waste Groups and Streams (continued)

Waste Stream	Symbol
Nuclear Power Plant Decommissioning (continued)	
PWR contaminated concrete	P-DECONCO
PWR combustible/compactible trash	P-DETRASH
PWR chelated ion-exchange resins	P-DERESIN
PWR filter cartridges	P-DEFILCR
PWR evaporator bottoms	P-DEEVAPB
BWR activated core shroud	B-DECORES
BWR activated reactor intervals	B-DEACINT
BWR activated reactor vessel	B-DEACVES
BWR activated concrete	B-DEACTCO
BWR contaminated metal	B-DECONME
BWR contaminated concrete	B-DECONCO
BWR combustible/compactible trash	B-DETRASH
BWR chelated ion-exchange resins	B-DERESIN
BWR evaporator bottoms	B-DEEVAPB
West Valley Demonstration Project	
Thorex high-level waste	W-THORHLW
Purex high-level waste	W-PUREHLW
Trash from existing systems	W-COTRASH
Miscellaneous dry solids	W-NCSOLID
LLWTF sludge and resins	W-LLWTFRE
FRS filter precoat and resins	W-FRSRESN
RTS liquid waste	W-FRSLIQD
RTS filter backwash and resins	W-RTSRESN
Trash, low TRU content	W-LTTRASH
Trash, high TRU content	W-HTTRASH
Equipment and hardware, low TRU	W-LTEQUIP
Equipment and hardware, high TRU	W-HTEQUIP
PD liquid waste	W-PDWLIQD
Vitrification supernate	W-VITSUPR
Vitrification sludge wash	W-VITWASH
Vitrification scrub condensate	W-VITSCRB
Vitrification melter feed overheads	W-VITMELT
Vitrification fractionator condensate	W-VITFRAC
Vitrification zeolite slurry	W-VITZEOL
D/D fuel storage racks	W-DDRACKS
D/D rubble, low TRU	W-DDLTRUB
D/D rubble, high TRU	W-DDHTRUB
D/D liquid, low TRU	W-DDLTLQD
D/D liquid, high TRU	W-DDHTLQD
D/D resins	W-DDRESIN
VII. Other Waste	
Spent nuclear power plant fuel rods	L-SPENTFU
Fuel assembly hardware	L-FUEHARD

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Table A-2 also contains the abbreviations used to identify each waste stream. These abbreviations are a shorthand way to represent the waste streams in further discussion, computer code data output, and so forth. Each waste stream begins with a single letter symbol which identifies the waste generator, and is followed by a seven-letter symbol that identifies the particular type of waste. The waste generator symbols correspond to the following:

Symbol	Waste Generator
P	Pressurized water reactors
B	Boiling water reactors
L	Light water reactors
F	Fuel fabrication facilities
U	UF_6 conversion plants
I	Institutional facilities
N	Industrial facilities
M	Military waste
R	Uranium recycle facilities
W	West Valley Demonstration Project

As shown, six of the waste streams have been separated into two components, and the additional waste streams resulting from this separation have been denoted by a plus sign after the waste generator symbol (I or N) instead of the usual minus sign. These streams are industrial SSTRASH, industrial LOTRASH, institutional COTRASH, institutional LIQSCLV, institutional ABSLIQD, and institutional BIOWAST. The reason for this separation is to identify the volumes of waste from generators that can more easily implement their own waste treatment processes (e.g., comparatively large facilities, denoted by a minus sign), and the waste from those generators that cannot do the same (e.g., comparatively small facilities, denoted by a plus sign). Waste volumes projected for each of these six waste streams are divided evenly between the two components. (The total volume of the industrial SSTRASH waste stream, for example, is the sum of the N+SSTRASH stream and the N-SSTRASH stream.)

In the nuclear fuel cycle waste group, the first seven waste streams are all "wet" wastes, and are occasionally referred to in this report as "process" wastes. Non-routine waste includes waste from nuclear power reactor decommissioning, nuclear fuel recycle (operation of a fuel reprocessing plant and mixed oxide fuel fabrication facility), Three Mile Island Unit 2 decontamination, and the West Valley Demonstration Project.

A.2 NUCLEAR POWER PLANTS

Central station nuclear power plants presently in operation in the United States include over 70 light water reactors (LWRs) and a single high temperature gas-cooled reactor (HTGR). The waste generated by the single HTGR is volumetrically and radiologically negligible compared with the wastes generated by LWRs, and is therefore not considered further in this appendix. Electricity for commercial use is also generated as a byproduct of the Hanford "N" plutonium production reactor. Wastes generated by this facility are disposed in facilities operated by the Department of Energy (DOE) and not in commercial disposal facilities. The Shippingport reactor was at one time operated by the Navy but is now scheduled for decommissioning as a demonstration project.

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A.2.1 Waste Generation Overview

The majority of the LWR waste streams are generated by operation of in-plant liquid radwaste processing systems. The basic functions of these processing systems are to reduce the accumulation of radioactive contaminants within the plant and to reduce the amount of these contaminants released from the plant. More detailed descriptions of these systems can be found elsewhere (Ref. 1). During these processes, radioactive contaminants are concentrated in several forms.

Two types of LWRs are in operation today: pressurized water reactors (PWRs) and boiling water reactors (BWRs). Waste streams common to PWRs and BWRs can be divided into "process wastes" and "trash." Process waste streams include ion-exchange resins, concentrated liquids (evaporator bottoms), and filter sludges. Cartridge filters are another form of process waste but are used much more extensively in PWRs than in BWRs. Trash waste streams can be divided into compactible trash and noncompactible trash. Another waste stream common to both types of LWRs and generated on an infrequent basis consists of nonfuel reactor core components. Wastes from future LWR decontamination operations may also be generated.

A brief discussion is given below of each of the waste streams characterized in this report as generated by nuclear power plants (Ref. 1).

Ion Exchange Resins

Processes involving ion exchange media are frequently used in LWRs to remove dissolved radioactivity from liquid streams. Ion exchange media usually consist of organic resins, which can be cation or anion resins, or a mixture of both. Inorganic zeolite ion exchange media have also been used in some cases. The resins (or other ion exchange media) are usually packed into cylindrical tanks (ion exchange columns or demineralizers) and the liquid containing the specific contaminant is passed through the resin column. In this process, dissolved radiocontaminants chemically displace ions in the resin and become physically bound to the resin. When an ion exchange bed can no longer perform its function (following depletion), it is replaced or regenerated. The old bed is typically transferred as a slurry out of the tanks into a shipping container (generally referred to as a liner), where excess water is removed prior to transfer to a disposal facility. Removal of free water is termed dewatering; dewatered ion exchange media, however, can still contain between 42 and 55% water by weight, in addition to interstitial liquid. In general, the liners are transported in casks that are shielded for radiation protection.

Concentrated Liquids

Concentrated liquid waste may be produced by the evaporation of a wide variety of LWR liquid streams. Many systems generating these liquid streams are interrelated. The waste consists of liquids with an elevated suspended and dissolved solids content, and also consists of sludge resulting from supersaturation during evaporation. Newer LWR plants, especially PWRs, often have several evaporators, each dedicated to concentrating a particular liquid stream. Existing PWRs usually concentrate boric acid waste solutions to about 11% solids by weight, and BWRs usually concentrate liquids containing sodium sulfate to about 25% solids by weight. Other types of solutions (e.g., laundry liquids,

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laboratory drains) are concentrated to about 25% solid by weight. These concentrated liquids are currently solidified in various matrix materials prior to transfer to a disposal facility.

Filter Sludge

Filter sludge is waste produced by precoat filters and consists of filter aid and waste solids retained by the filter aid. Diatomaceous earth, powdered mixtures of cation and anion exchange resins, and high purity cellulose fibers are common filter aids. These materials are slurried and deposited (precoated) as a thin cake on the initial filter medium (wire mesh, cloth, etc.). The filter cake removes suspended solids from liquid streams. Precoat filters using powdered resins also remove dissolved solids but are not as effective as mixed bed ion exchange columns (deep bed demineralizers) due to the shorter contact time of the liquid with the resin. Precoat filters are used much more extensively in BWRs than in PWRs. Precoat filtration may be used in conjunction with ion exchange columns and evaporation, or it may be the only form of treatment removing suspended solids from a particular liquid stream.

Cartridge Filters

Cartridge filters contain one or more disposable filter elements. These elements may be typically constructed of woven fabric, wound fabric, or pleated paper supported internally by a stainless steel mesh as well as pleated or matted paper supported by an external stainless steel basket. Paper filter elements are often impregnated with epoxy. Woven fabric filters are typically constructed of cotton and nylon. Cartridge filters are effective in removing suspended solids, but do not have the ion exchange capability of precoat filters or demineralizers. They are used much more extensively in PWRs than BWRs, and their typical uses in LWRs are similar to those of precoat filters. Many plants use cartridge filters in conjunction with ion exchange columns, evaporators, and precoat filters. These cartridge filters are frequently packed in 55-gallon drums (between 3 to 12 per drum) prior to transfer to a disposal facility.

Trash

Trash is the most varied waste stream generated by LWRs and can contain everything from paper towels to irradiated reactor internals. Some of the materials that have been identified in the past as having been shipped as trash are listed in Table A-3.

Compactible and noncompactible items are frequently shipped in the same container; in addition, packaging small pieces of activated metal with relatively innocuous materials is common. Such factors make characterization of trash difficult. In general, compactible trash contains more combustible material (e.g., paper, plastic), and noncompactible trash contains more metallic components (e.g., pipes and failed equipment). It is usually assumed that the volume percentage of compactible trash and combustible trash are the same. Similarly, the volume percentages of noncompactible trash and noncombustible trash are assumed to be the same.

Other Waste Streams

Nonfuel reactor components consist of fuel channels, control rods, control rod channels, shim rods, in-core instrumentation, and flux wires. Many of these

Table A-3. Material Shipped as LWR Trash

Material	BWRs		PWRs	
	C*	N*	C	N
Anti-contaminant clothing			X	
Cloth (rags, mops, gloves)	X		X	
Conduit				X
Contaminated dirt	X	X		
Contaminated tools and equipment				
Hand tools	X	X		
Eddy current equipment				X
Vessel inspection equipment				X
Ladders		X		X
Lighting fixtures				X
Spent fuel racks				X
Scaffolding		X	X	
Laboratory equipment			X	X
Filters				
Filter cartridges	X	X		
HEPA filters		X	X	X
Respirator cartridges	X			
Glass	X	X	X	
Irradiated Metals				X
Flux wires				X
Flow channels		X		
Fuel channels		X		X
In-core instrumentation				X
Poison channels		X		
Shim rods				X
High density concrete block				X
Miscellaneous metal	X	X		X
Aerosol cans		X		
Buckets	X			
Crushed 55-gal drums		X		
Fitting		X		
Pipes and valves		X		X
Miscellaneous wood	X	X	X	X
Paper	X		X	
Plastic				
Bags, gloves, shoe covers	X	X	X	
Sample bottles	X			
Rubber	X		X	
Sweeping compounds		X		

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*C: compactible, combustible;
N: noncompactible, noncombustible.

components are exposed to the primary reactor coolant and all are exposed to the in-core neutron flux.

LWR decontamination waste is expected to be produced in the future as part of full-scale decontamination of LWR primary coolant systems. The purpose of decontamination is to reduce in-plant occupational radiation exposures by removing crud accumulated on surfaces that are in contact with the primary coolant. It is expected that typical waste streams generated during these future routine decontamination operations will include such streams as ion-exchange resins used to process the decontamination solutions and solidified evaporator bottoms. The wastes are projected to contain large quantities of chelating agents.

A.2.2 Waste Volume Projections

Similar to reference 1, the approach used to project nuclear power plant waste volumes is to first make a projection of the nuclear electrical generation capacity. Then, the waste volumes are assumed to be proportional to the capacity.

A.2.2.1 Electrical Generation Capacity

Projections of power generation rates for nuclear power were made in Appendix A of reference 1 on the basis of a review of nuclear power stations currently built and operable, under construction, or planned or on order. Two scenarios were assumed in Appendix A for nuclear power station construction, a "low scenario" and a "high scenario." The high scenario resulted in a total projected capacity by the year 2000 of 169,141 MW(e), which was used for purposes of calculating impacts from waste disposal in the draft and final Part 61 EIS (Refs. 3, 4).

Since these projections were originally made, however, a number of plants have been canceled, including several which were already under construction. In other instances, construction has been deferred indefinitely. Thus, an updated projection of electrical capacity is needed.

Table A-4 lists by NRC region the nuclear power stations assumed to be operable in 1984, and represents an update of Table 4 in Appendix A of reference 1. The capacities and startup dates were obtained principally from reference 9. In this table (and also in Tables A-5 through A-7), light water reactors are identified by type (PWR or BWR) and by other distinguishing characteristics which influence generation of waste. These include the following as noted in reference 10:

Identifier	Description
1	PWR, fresh water cooling
2	PWR, salt water cooling
3	BWR, fresh water cooling, deep bed condensate polishing system (CPS)
4	BWR, fresh water cooling, filter/demineralizer CPS
5	BWR, salt water cooling

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Table A-4. Nuclear Power Reactors Assumed to be in Operation in 1984

Reactor	State Located	Type	Capacity MW(e)	Startup
Region 1				
Beaver Valley 1	PA	PWR1	852	1976
Calvert Cliffs 1	MD	PWR2	845	1974
Calvert Cliffs 2	MD	PWR2	845	1974
Indian Point 2	NY	PWR2	873	1973
Indian Point 3	NY	PWR2	965	1976
Fitzpatrick	NY	BWR3	821	1974
Haddam Neck (Conn. Yankee)	CT	PWR1	575	1967
Maine Yankee	ME	PWR2	825	1972
Millstone 1	CT	BWR5	660	1970
Millstone 2	CT	PWR2	870	1975
Nine Mile Point 1	NY	BWR3	620	1969
Oyster Creek 1	NJ	BWR5	650	1969
Peach Bottom 2	PA	BWR4	1065	1973
Peach Bottom 3	PA	BWR4	1065	1974
Pilgrim 1	MA	BWR5	655	1974
R. E. Ginna 1	NY	PWR1	470	1969
Salem 1	NJ	PWR2	1090	1976
Salem 2	NJ	PWR2	1115	1981
Susquehanna 1	PA	BWR4	1050	1983
Vermont Yankee	VT	BWR4	514	1972
Yankee-Rowe	MA	PWR1	175	1960
Region 2				
Browns Ferry 1	AL	BWR4	1065	1973
Browns Ferry 2	AL	BWR4	1065	1974
Browns Ferry 3	AL	BWR4	1065	1976
Brunswick 1	NC	BWR5	821	1976
Brunswick 2	NC	BWR5	821	1975
Crystal River 3	FL	PWR2	825	1977
E. I. Hatch 1	GA	BWR4	786	1974
E. I. Hatch 2	GA	BWR4	784	1978
H. B. Robinson	SC	PWR1	700	1970
J. M. Farley 1	AL	PWR1	829	1977
J. M. Farley 2	AL	PWR1	829	1981
North Anna 1	VA	PWR1	907	1978
North Anna 2	VA	PWR1	907	1980

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Table A-4. (Continued)

Reactor	State Located	Type	Capacity MW(e)	Startup
Region 2 (continued)				
Oconee 1	SC	PWR1	887	1973
Oconee 2	SC	PWR1	887	1973
Oconee 3	SC	PWR1	887	1974
St. Lucie 1	FL	PWR2	802	1976
St. Lucie 2	FL	PWR2	810	1983
Sequoyah 1	TN	PWR1	1148	1981
Sequoyah 2	TN	PWR1	1148	1982
Surry 1	VA	PWR2	822	1972
Surry 2	VA	PWR2	822	1973
Turkey Point 3	FL	PWR2	693	1972
Turkey Point 4	FL	PWR2	693	1973
V. C. Summer 1	SC	PWR1	900	1984
W. B. McGuire 1	NC	PWR1	1180	1981
W. B. McGuire 2	NC	PWR1	1180	1984
Region 3				
Big Rock Point	MI	BWR3	72	1962
Davis-Besse 1	OH	PWR1	906	1977
D. C. Cook 1	MI	PWR1	1054	1975
D. C. Cook 2	MI	PWR1	1100	1978
Dresden 2	IL	BWR3	794	1970
Dresden 3	IL	BWR3	794	1971
Duane Arnold 1	IA	BWR4	538	1974
Kewannee	WI	PWR1	535	1974
La Crosse (Genoa)	WI	BWR3	50	1967
La Salle 1	IL	BWR3	1078	1982
La Salle 2	IL	BWR3	1078	1984
Monticello	MN	BWR4	545	1970
Palisades	MI	PWR1	805	1971
Point Beach 1	WI	PWR1	497	1970
Point Beach 2	WI	PWR1	497	1972
Prairie Island 1	MN	PWR1	530	1973
Prairie Island 2	MN	PWR1	530	1974
Quad-Cities 1	IL	BWR4	789	1972
Quad-Cities 2	IL	BWR4	789	1972
Zion 1	IL	PWR1	1040	1973
Zion 2	IL	PWR1	1040	1973
Region 4				
Arkansas 1	AR	PWR1	850	1974
Arkansas 2	AR	PWR1	912	1980
Cooper	NE	BWR4	778	1974

Table A-4. (Continued)

Reactor	State Located	Type	Capacity MW(e)	Startup
Region 4 (continued)				
Ft. Calhoun	NE	PWR1	457	1973
Ft. St. Vrain	CO	HTGR	330	1974±
Region 5				
Rancho Seco 1	CA	PWR1	918	1974
San Onofre 1	CA	PWR2	436	1967
San Onofre 2	CA	PWR2	1100	1983
San Onofre 3	CA	PWR2	1100	1984
Trojan 1	OR	PWR1	1130	1975
WNP-2	WA	BWR4	1100	1984

±The Fort St. Vrain high temperature graphite reactor generates, compared to a light water reactor, a negligible quantity of low-level waste.

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Table A-5 is a listing by NRC region of nuclear power stations currently under construction and for which an expected startup date has been listed in reference 11. Table A-5 is essentially an update of Table 5 in Appendix A of reference 1. (The projected startup dates shown in Table A-5 are those provided by reference 11 while the capacities remain those given in reference 9.)

Reference 11 also lists a number of other nuclear power stations as having "indefinite" startup times, and these stations are listed in Table A-6. The startup dates for these units, which is believed to be problematical for some of them, are difficult to project. For example, WNP-3 has been put on hold by the utility due to lack of funds (Ref. 12).

Table A-7 lists a number of units which are not currently operating but which generate waste as a result of routine maintenance operations, spent fuel storage, and so forth. The Indian Point 1 and Humboldt Bay units are being decommissioned. These units are being held in safe storage pending dismantlement, which will probably not be until the approximate end of this century. The Dresden 1 unit has been shut down for a number of years. The plant operator, Commonwealth Edison, did not originally plan to make a decision on restarting the plant until June 1986. The plant operators were uncertain whether it would be economically worthwhile to make a number of modifications imposed by NRC since the accident at Three Mile Island (TMI). More recent indications, however, are that Commonwealth Edison plans to "mothball" the plant until Units 2 and 3 are eventually decommissioned (Ref. 13). Both units at Three Mile Island have been shut down since the March 1979 accident at Unit 2.

Over the interim, the units listed in Table A-7 are assumed to generate waste proportional to a fictitious electrical power rating of 50 MW(e). (100 MW(e) is assumed for Three Mile Island, Unit 2.) Some units are conservatively postulated to be restarted according to the schedule listed in the table. These startup dates are uncertain and are only included to ensure a conservative radioactive waste source term. At the time of restart, these units are assumed to generate waste proportional to the rated electrical generating capacity.

For the purposes of developing waste projections, the units listed as operable in Table A-4 are assumed to operate (and generate waste) at the listed electrical capacity until they are decommissioned. Additional units are assumed to come on line as projected in Table A-5 and also operate. Units listed in Table A-6 are also assumed to come on line as shown. These assumed startup times in Table A-6 are little more than educated guesses, and have been included to ensure a conservative source term. Similarly, units which are currently shut down are conservatively assumed to be restarted according to the schedule given in Table A-7.

The resulting annual additions to the light water reactor electrical capacity are listed in Tables A-8 and A-9 by year, by NRC region, and by type of power plant from 1980 to the year 2030. The combined total (including TMI-1 and 2 but not Indian Point 1, Dresden 1, or Humboldt Bay) by the year 2000 comes to about 117,000 MW(e), which is about 52,000 MW(e) less than the electrical capacity assumed in the draft Part 61 EIS (Ref. 3). This revised figure is still believed to be conservative, since it is believed likely that several of the listed plants in Tables A-5 and A-6 will be canceled or delayed. For example, progress on Limerick 2 and Seabrook 2 has been slowed considerably and one or both of these units may be canceled (Refs. 12, 14). Seabrook 2 has in fact been

Table A-5. Nuclear Power Generating Units Under Construction in 1984

Reactor	State Located	Type	Capacity MW(e)	Assumed Startup
Region 1				
Beaver Valley 2	PA	PWR1	833	1986
Hope Creek 1	NJ	BWR5	1067	1986
Limerick 1	PA	BWR4	1065	1985
Limerick 2	PA	BWR5	1065	1988
Millstone 3	CT	PWR2	1156	1986
Nine Mile Point 2	NY	BWR4	1100	1986
Seabrook 1	NH	PWR2	1200	1986
Shoreham	NY	BWR5	819	1985
Susquehanna 2	PA	BWR4	1050	1985
Region 2				
A. W. Vogtle 1	GA	PWR1	1110	1987
A. W. Vogtle 2	GA	PWR1	1100	1988
Bellefonte 1	AL	PWR1	1213	1989
Bellefonte 2	AL	PWR1	1213	1991
Catawba 1	SC	PWR1	1145	1985
Catawba 2	SC	PWR1	1145	1987
Grand Gulf 1	MS	BWR3	1250	1985
River Bend 1	LA	BWR3	934	1985
Shearon Harris 1	NC	PWR1	900	1986
Waterford 3	LA	PWR1	1113	1985
Watts Bar 1	TN	PWR1	1177	1985
Watts Bar 2	TN	PWR1	1177	1988

Table A-5. (Continued)

Reactor	State Located	Type	Capacity MW(e)	Assumed Startup
Region 3				
Braidwood 1	IL	PWR1	1120	1986
Braidwood 2	IL	PWR1	1120	1987
Byron 1	IL	PWR1	1120	1985
Byron 2	IL	PWR1	1120	1986
Callaway 1	MO	PWR1	1120	1985
Carroll County 1	IL	PWR1	1120	2001
Carroll County 2	IL	PWR1	1120	2002
Clinton 1	IL	BWR4	933	1986
E. Fermi 2	MI	BWR4	1093	1985
Perry 1	OH	BWR3	1205	1985
Region 4				
Comanche Peak 1	TX	PWR1	1111	1985
Comanche Peak 2	TX	PWR1	1111	1986
South Texas 1	TX	PWR2	1250	1987
South Texas 2	TX	PWR2	1250	1989
Wolf Creek	KS	PWR1	1150	1985
Region 5				
Diablo Canyon 1	CA	PWR2	1084	1985
Diablo Canyon 2	CA	PWR2	1106	1985
Palo Verde 1	AZ	PWR1	1270	1985
Palo Verde 2	AZ	PWR1	1270	1986
Palo Verde 3	AZ	PWR1	1270	1987

Table A-6. Nuclear Power Reactors Having Indefinite Startup Dates as of 2/85

Plant	State Located	Type	Capacity Mw(e)	Projected Startup as of		% Const. (2/85)#	Assumed Startup
				1980*	8/81**		
Region 2							
Grand Gulf 2	MS	BWR4	1250	1985	1986	33	1989
Seabrook 2	NH	PWR2	1200	1986		23	1992
Region 3							
Perry 2	OH	BWR4	1205	1987	1988	44	1990
Region 5							
WNP 1	WA	PWR1	1218	1984	1986	62.5	1989
WNP 3	WA	PWR2	1242	1984	1986	75	1989

*Ref. 1
**Ref. 16
#Ref. 11

Table A-7. Reactor Units Currently Inoperable

Reactor	Region	State	Type	Capacity (MWe)	Postulated Restart
Dresden 1	3	IL	BWR3	200	-
Humboldt Bay 1	5	CA	BWR5	65	-
Indian Point 1	1	NY	BWR2	265	-
Three Mile Island 1	1	PA	PWR1	819	1986*
Three Mile Island 2	1	PA	PWR1	906	1990

*This reactor went critical in 1985. Full power operation is not anticipated until at least 1986.

Table A-8. Annual Additions to Assumed PWR Electrical Generating Capacity (MW(e))

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
1980	Beaver Valley 1	852	Crystal Rvr 3	825	Davis-Besse 1	906	Arkansas 1	850	Rancho Seco	918
	Calvert Clffs 1	845	H B Robinson	700	D C Cook 1	1054	Arkansas 2	912	San Onofre 1	436
	Calvert Clffs 2	845	J M Farley 1	829	D C Cook 2	1100	Ft Calhoun	457	Trojan 1	1130
	Indian Pt 1	50*	North Anna 1	907	Kewannee	535		2219		2484
	Indian Pt 2	873	North Anna 2	907	Palisades	805				
	Indian Pt 3	965	Oconee 1	887	Point Beach 1	497				
	Haddam Neck	575	Oconee 2	887	Point Beach 2	497				
	Maine Yankee	825	Oconee 3	887	Prairie Isl 1	530				
	Millstone 2	830	St. Lucie 1	802	Prairie Isl 2	530				
	R E Ginna 1	470	Surry 1	822	Zion 1	1040				
	Salem 1	1090	Surry 2	822	Zion 2	1040				
	Yankee Rowe	175	Turkey Pt 3	693		8534				
	TMI-1	50*	Turkey Pt 4	693						
	TMI-2	100*		10661						
		8545								
1981	Salem 2	1115	J M Farley 2	829						
			Sequoyah 1	1148						
			W B McGuire 1	1180						
				3157						
1982			Sequoyah 2	1148						
1983			St. Lucie 2	810					San Onofre 2	1100
1984			V C Summer 1	900						
			W B McGuire 2	1180					San Onofre 3	1100
				2080						
1985			Catawba 1	1145	Byron 1	1120	Comanche Pk 1	1111	Diablo Can 1	1084
			Waterford 3	1113	Callaway 1	1120	Wolf Creek	1150	Diablo Can 2	1106
			Watts Bar 1	1177		2240		2261	Palo Verde 1	3460
				3435						

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Table A-8. (Continued)

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
1986	Beaver Valley 2	833	S Harris 1	900	Braidwood 1	1120	Comanche Pk 2	1111	Palo Verde 2	1270
	Millstone 3	1156			Byron 2	1120				
	Seabrook 1	1200				2240				
	TMI-1	(-50)								
	TMI-1	819								
		3958								
1987			A W Vogtle 1	1110	Braidwood 2	1120	S Texas 1	1250	Palo Verde 3	1270
			Catawba 2	1145						
				2255						
1988			A W Vogtle 2	1100						
			Watts Bar 2	1177						
				2277						
1989			Bellefonte 1	1213			S Texas 2	1250	WNP 1	1218
									WNP 3	1242
										2460
1990	TMI-2	(-100)								
	TMI-2	906								
		806								
1991			Bellefonte 2	1213						
1992	Seabrook 2	1200								
1993										
1994										
1995										
1996										
1997										

Table A-8. (Continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
1998					
1999					
2000	Yankee Rowe (-175)				
2001			Carroll Cty 1 1120		
2002			Carroll Cty 2 1120		
2003					
2004					
2005					
2006					
2007	Haddam Neck (-575)				San Onofre 1 (-436)
2008					
2009	R E Ginna (-470)				
2010		H B Robinson (-700)	Pt Beach 1 (-497)		
2011			Palisades (-805)		
2012	Maine Yankee (-825)	Surry 1 (-822) Turkey Pt 3 (-693) (-1515)	Pt Beach 2 (-497)		

Table A-8. (Continued)

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
2013	Indian Pt 1	(- 50)	Oconee 1	(-887)	Prairie Isl 1	(-530)	Ft. Calhoun	(-457)		
	Indian Pt 2	(-873)	Oconee 2	(-887)	Zion 1	(-1040)				
		(-923)	Surry 2	(-822)	Zion 2	(-1040)				
			Turkey Pt 4	(-693)		(-2610)				
				(-3289)						
2014	Cal Cliffs 1	(-845)	Oconee 3	(-887)	Kewaunee	(-535)	Arkansas 1	(-850)	Rancho Seco	(-918)
	Cal Cliffs 2	(-845)			Prairie Isl 2	(-530)				
	TMI-1	(-819)				(-1065)				
		(-2509)								
2015	Millstone 2	(-830)			D C Cook 1	(-1054)			Trojan 1	(-1130)
2016	Beaver Val 1	(-852)	St Lucie 1	(-802)						
	Indian Pt 3	(-965)								
	Salem 1	(-1090)								
		(-2907)								
2017			Crystal Rvr 3	(-825)	Davis-Besse 1	(-906)				
			J M Farley 1	(-829)						
				(-1654)						
2018			North Anna 1	(-907)	D C Cook 2	(-1100)				
2019	TMI-2	(-906)								
2020			North Anna 2	(-907)			Arkansas 2	(-912)		
2021	Salem 2	(-1115)	J M Farley 2	(-829)						
			Sequoyah 1	(-1148)						
			WB McGuire 1	(-1180)						
				(-3157)						

Table A-8. (Continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
2022		Sequoyah 2 (-1148)			
2023		St Lucie 2 (-810)			S Onofre 2 (-1100)
2024		V C Summer (-900) W B McGuire (-1180) (-2080)			Diab Can 1 (-1084) S Onofre 3 (-1100) (-2184)
2025		Catawba 1 (-1145) Waterford 3 (-1113) Watts Bar 1 (-1177) (-3435)	Byron 1 (-1120) Callaway 1 (-1120) (-2240)	Com Pk 1 (-1111) Wolf Creek (-1150) (-2261)	Palo Verde 1(-1270) Diab Can 1 (-1084) Diab Can 2 (-1106) (-3460)
2026	Beaver Val 2 (-833) Millstone 3 (-1156) Seabrook 1 (-1200) (-3189)	S Harris 1 (-900)	Braidwood 1 (-1120) Byron 2 (-1120) (-2240)	Com Pk 1 (-1111)	Palo Verde 2(-1270)
2027		A W Vogtle 1 (-1110) Catawba 2 (-1145) (-2255)	Braidwood 2 (-1120)	S Texas 1 (-1250)	Palo Verde 3(-1270)
2028		A W Vogtle 2 (-1110) Watts Bar 2 (-1177) (-2277)			
2029		Bellefonte 1 (-1213)		S Texas 2 (-1250)	WNP 1 (-1218) WNP 3 (-1242) (-2460)

Table A-8. (Continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
2030		Bellefonte 2 (-1213)			
2031					
2032	Seabrook 2 (-1200)				
2033					
2034					
2035					
2036					
2037					
2038					
2039					
2040					
2041			Carroll Cty 1 (-1120)		
2042			Carroll Cty 2 (-1120)		

Table A-9. Annual Additions to BWR Electrical Generating Capacity (MW(e))

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
1980	Fitzpatrick	821	Browns Ferry 1	1065	Big Rock Pt	72	Cooper	778	Humboldt Bay	50*
	Millstone 1	660	Browns Ferry 2	1065	Dresden 1	50*				
	Nine Mile Pt 1	620	Browns Ferry 3	1065	Dresden 2	794				
	Oyster Creek	650	Brunswick 1	821	Dresden 3	794				
	Peach Bott 2	1065	Brunswick 2	821	Duane Arnold	538				
	Peach Bott 3	1065	E I Hatch 1	786	La Crosse	50				
	Pilgrim	655	E I Hatch 2	784	Monticello	545				
	Vermont Yankee	514		6407	Quad-Cities 1	1040				
		6050			Quad-Cities 2	1040				
						4923				
1981										
1982					La Salle 1	1078				
1983	Susquehanna 1	1050								
1984					La Salle 2	1078			WNP 2	1100
1985	Limerick 1	1065	Grand Gulf 1	1250	E Fermi 2	1093				
	Shoreham	819	River Bend 1	934	Perry 1	1205				
	Susquehanna 2	1050		2184		2298				
		2934								
1986	Hope Creek 1	1067			Clinton 1	933				
	Nine Mile Pt 2	1100								
		2167								
1987										
1988	Limerick 2	1065								
1989			Grand Gulf 2	1250						

Table A-9. (Continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
1990			Perry 2 1205		
1991					
1992					
1993					
1994					
1995					
1996					
1997					
1998					
1999					
2000					Humboldt Bay (- 50)
2001					
2002			Big Rock Pt (-72)		
2003					
2004					
2005					
2006					

Table A-9. (Continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
2007			La Crosse (-50)		
2008					
2009	Oyster Creek (-650) Nine Mi Pt 1 (-620) (-1270)				
2010	Millstone 1 (-660)		Dresden 2 (-744) Monticello (-545) (-1339)		
2011			Dresden 1 (-50) Dresden 2 (-50) Dresden 3 (-794) (-894)		
2012	Vt Yankee (-514)		Qd Cities 1 (-1040) Qd Cities 2 (-1040) (-2080)		
2013	Peach Bott 2 (-1065)		Brns Ferry 1 (-1065)		
2014	Fitzpatrick (-821) Peach Bott 3 (-1065) Pilgrim (-655) (-2541)	Brns Ferry 2 (-1065) E I Hatch 1 (-786) (-1851)	Duane Arnold (-538)	Cooper (-778)	
2015		Brunswick 2 (-821)			
2016		Brns Ferry 3 (-1065) Brunswick 1 (-821) (-1886)			

Table A-9. (Continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
2017					
2018		E I Hatch 2 (-784)			
2019					
2020					
2021					
2022			La Salle 1 (-1078)		
2023	Susquehanna 1 (-1050)				
2024			La Salle 2 (-1078)		WNP-2 (-1100)
2025	Limerick 1 (-1065) Susquehanna 2 (-1050) Shoreham (-819) (-2934)	Grand Gulf 1 (-1250) River Bend 1 (-934) (-2184)	E Fermi 2 (-1093) Perry 1 (-1205) (-2298)		
2026	Hope Creek 1 (-1067) Nine Mi Pt 2 (-1100)		Clinton 1 (-933)		
2027					
2028	Limerick 2 (-1065)				
2029		Grand Gulf 2 (-1250)			
2030			Perry 2 (-1205)		

placed on hold pending completion of Unit 1, although a startup date is nonetheless assumed in Table A-6. Planning for the two Carroll County plants is also indefinite, and the utility, Commonwealth Edison, has not even applied to NRC for construction permits. At this time there are no light water reactor projects seeking construction permits from NRC (Ref. 12).

The projections include an assumed decommissioning schedule in which units are assumed to be shut down and dismantled 40 years following startup. Thus, the electrical generating capacity for the shut down unit is subtracted at the appropriate time period. Units currently shut down are assumed to be held in storage until the year 2000, the approximate time in which a Federal repository is projected to be made available for storage or disposal of spent reactor fuel (see Section A.7.2).

A.2.2.2 Principal Light Water Reactor Waste Streams

Background

When reference 1 and the draft Part 61 EIS (Ref. 3) were prepared, there was little available data on the comparative generation rates of different wastes streams from light water reactors. One of the few sources of data was ONWI-20 (Ref. 15), and based on the ONWI-20 data the following table was prepared and used:

	Volumes (m^3/MW(e)-yr)		Activity (Ci/MW(e)-yr)	
Waste Type	BWR	PWR	BWR	PWR
A. Wet				
Resins	0.081	0.018	1.14	0.40
Concentrated liquids	0.223	0.124	0.20	0.11
Filter sludge	0.179	0.002	1.40	0.006
Cartridge filters	0	0.011	0	0.12
Total:	0.483	0.155	2.74	0.636
B. Dry (trash)				
Compactible	0.221	0.215	0.005	0.005
Noncompactible	0.105	0.111	0.397	0.058
Total:	0.326	0.326	0.402	0.063
Totals:	0.808	0.478	3.29	0.699

For the table, 60% of BWRs were assumed to use deep-bed condensate polishing systems (CPS) and 40% precoat CPS; about half of PWRs were assumed to use CPS in their secondary coolant loops and about half were not. The volumes shown above, with the exception of cartridge filters, are for "untreated" wastes. Concentrated liquids (evaporator bottoms) are reported as generated prior to solidification. Resins and filter sludges are reported as dewatered, and the trash streams are reported as generated prior to such processing options as incineration or compaction. The volumes for cartridge filters are given as packaged for shipment.

More recent information indicates that the ONWI-20 data (Ref. 15) may underestimate volumes of some waste streams. One source of information consists of reports provided semi-annually to NRC by power reactors of radioactive material released in the form of effluents and waste shipments. These semi-annual release reports are periodically summarized and published by NRC, and three such published reports for the years 1978, 1979, and 1980 have been considered in this appendix (Refs. 17-19). Also considered were semi-annual release reports for the years 1981 and 1982, as obtained from the Department of Energy (DOE) Low-Level Waste Management Information System. In these semi-annual reports, the volumes and activities of waste shipped to LLW disposal sites are listed for each power reactor facility. The waste volumes and activities are furthermore generally divided into four categories: wet wastes (spent resins, filter sludges, evaporation bottoms, etc.), dry wastes (dry compressible waste, contaminated equipment, etc.), irradiated components, and other wastes. Occasionally, however, specific categories of waste are not (or are incompletely) specified.

To use the data, the reports provided for each reactor facility were examined, and some records discarded. Records discarded included those for facilities for which the waste categories were unspecified, those for which the information was otherwise suspect, and those for facilities reporting waste from non-operating reactor facilities. The remaining waste volumes and activities in each category were then summed for each year. Also summed was the rated capacity for each power plant using the capacities listed in Table A-4. The resulting summations are presented in Table A-10, and also imply the following (as-shipped) waste generation rates for wet and dry wastes:

Plant	Wet Wastes		Dry Wastes	
Type	m^3/MW(e)-yr	Ci/MW(e)-yr	m^3/MW(e)-yr	Ci/MW(e)-yr
PWR	0.363	0.466	0.369	0.047
BWR	0.422	1.722	0.838	0.084

Table A-10. Summary of Information from Semi-Annual Release Reports

Year	Reactor Type	Capacity (MW(e)-yr)	Wet Wastes		Dry Wastes		Irradiated Comp.		Other	
			m^3	Ci	m^3	Ci	m^3	Ci	m^3	Ci
1978	P	17,063.	4,154.65	6,336.63	3,818.71	666.92	383.83	45,392.68	94.	492.95
	B	11,722.	4,976.34	13,452.6	7,603.1	580.33	194.65	272,207.	803.8	2.76
1979	P	19,106.	17,985.76	5,454.23	6,349.2	530.88	102.2	4,599.45	265.81	235.98
	B	11,722.	5,262.	17,543.1	7,267.9	744.18	2,017.75	32,155.97	772.1	24.5
1980	P	23,220.	5,190.4	16,972.96	9,616.5	1,382.52	373.2	14,319.4	3,076.11	32.73
	B	11,722.	6,119.	24,427.3	13,684.6	1,068.05	662.5	21,403.53	0	0
1981	P	19,183.	3,445.08	10,133.99	5,845.8	565.62	178.47	15,611.56	249.45	66.18
	B	11,476.	4,965.31	25,402.99	10,980.12	1,177.67	6.86	87.	625.	50.15
1982	P	19,274.	4,739.74	6,688.21	10,429.73	1,469.81	397.97	1,852.35	434.37	383.15
	B	11,722.	3,291.25	19,660.42	9,396.14	1,313.35	39.7	9,203.04	4,448.7	488.71
Totals:	P	97,846.	35,515.63	45,586.02	36,059.94	4,615.75	1,435.67	81,775.44	4,119.74	1,210.99
	B	58,364.	24,613.9	100,486.41	48,931.86	4,883.58	2,921.46	335,056.54	6,649.6	566.12

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The old ONWI-20 data, however, implied the following as-shipped generation rates, assuming that concentrated liquids have been solidified (volume increase factor = 1.4), that compactible wastes have been compacted (volume reduction factor = 2), and that resins and filter sludge are shipped in a dewatered form:

	Volumes (m^3/MW(e)-yr)	
Waste Type	PWR	BWR
A. Wet		
Resins	0.018	0.081
Conc. liqs.	0.174	0.312
Filter Sludge	0.002	0.179
Cart. Filters	0.011	0
Total:	0.205	0.572
B. Dry		
Compactible	0.108	0.111
Noncompactible	0.111	0.105
Total:	0.219	0.216

As can be seen, the ONWI-20 data seems to underestimate the volumes of PWR wet wastes, PWR dry wastes, and BWR dry wastes, while overestimating the volumes of BWR wet wastes.

Another source of information is reference 10. This study obtained a considerable amount of information related to waste sources and operational characteristics from a number of operating nuclear power plants. One of the conclusions of this study was that waste generation at an actual operating nuclear power plant did not correlate well with the plant's electrical capacity. Rather, the study concluded that radioactive waste generation rates were a function of several operational parameters, and that no one parameter correlates directly with either wet or dry waste production. In terms of dry waste generation, the parameters that seemed to show the best correlation at PWRs were the number of personnel onsite with measurable exposure and the total personnel exposure. For BWRs, dry waste generation seemed to correlate with outage duration. Wet waste generation at both PWRs and BWRs seemed to correlate with dry waste generation.

Waste Projections

Principal light water waste streams and volume generation rates are listed in Table A-11 as a function of reactor type (PWR or BWR) and other distinguishing characteristics which influence generation of waste. These include:

Abbreviation	Description
PWR-F	PWR, fresh water cooling
PWR-S	PWR, salt water cooling
BWR-FDB	BWR, fresh water cooling, deep bed condensate polishing system (CPS)
BWR-FFD	BWR, fresh water cooling, filter/ demineralizer CPS
BWR-S	BWR, salt water cooling

Volume projection rates are made based principally on data presented in reference 10, and are given in units of m^3 per annual electrical capacity. This is done for convenience, although in view of the information presented in reference 10, additional work is probably needed to develop improved predictive ability.

In Table A-11, resin and filter sludge volumes are given as dewatered, while concentrated liquid volumes are given prior to solidification. Cartridge filter volumes are given as packaged. Trash waste stream volumes are given as-generated (prior to further processing such as compaction).

Table A-11. Principal Light Water Reactor Waste Streams and Generation Rates

Waste Streams		Generation Rates (m³/MW(e)-yr)					
Symbol	Description	PWR-F	PWR-S	BWR-FDB	BWR-FFD	BWR-S	Gross Concentration (Ci/m³)
P-IXRESIN	PWR ion-exchange resins	2.24E-2	6.37E-2	-	-	-	*
P-CONCLIQ	PWR concentrated liquids	1.77E-1	8.67E-2	-	-	-	*
P-FSLUDGE	PWR filter sludge	-	-	-	-	-	*
P-FCARTRG	PWR filter cartridges	7.36E-3	1.08E-2	-	-	-	4.48E+0
B-IXRESIN	BWR ion-exchange resins	-	-	4.90E-2	3.82E-2	1.73E-1	*
B-CONCLIQ	BWR concentrated liquids	-	-	3.25E-1	-	3.78E-1	*
B-FSLUDGE	BWR filter sludge	-	-	3.40E-2	2.23E-1	2.64E-1	*
P-COTRASH	PWR compressible trash	1.64E-1	1.97E-1	-	-	-	*
P-NCTRASH	PWR noncompressible trash	1.55E-1	2.74E-1	-	-	-	*
B-COTRASH	BWR compressible trash	-	-	2.87E-1	3.98E-1	9.87E-1	*
B-NCTRASH	BWR noncompressible trash	-	-	7.00E-2	2.76E-1	5.09E-1	*

*Concentrations for all waste streams except PWR cartridge filters are given as weighted averages across waste stream specific concentration distributions. Insufficient data is available to do this for PWR cartridge filters, and so concentrations for PWR cartridge filters are given as an average across the entire waste stream.

For filter cartridges, an average gross concentration of 4.48 Ci/m^3 is assumed. For the remaining waste streams, however, gross concentrations are given as distributions across the volumes of the streams. This approach was used in the final Part 61 EIS (Ref. 4), and the assumed concentration distributions are given in Tables A-12 and A-13. Concentration distributions for the LWR process waste streams were obtained from the final Part 61 EIS. Concentration distributions for the LWR trash waste streams (Table A-13) were estimated based on information obtained from waste generation records from nuclear power plants (Ref. 20). In this study, the following distributions were compiled from shipments of dry active wastes from several nuclear power plants; where the concentration range and the average concentration within the range are in units of Ci/m^3:

PWR Comp. Trash			PWR Noncomp. Trash		
Range	%	Average	Range	%	Average
0-.0411	26	.0296	0-.0411	49	.0208
→.0823	23	.0665	→.0822	23	.0716
→.329	35	.266	→.329	20	.238
→.823	12	.542	→.822	-	-
→4.11	4	1.35	→4.11	8	3.79

BWR Comp. Trash			BWR Noncomp. Trash		
Range	%	Average	Range	%	Average
0-.0292	70	.0181	0-.0295	69	.0176
→.0584	13	.0520	→.0590	9	.0527
→.233	11	.126	→.236	15	.130
→.584	4	.406	→.590	4.7	.415
→2.92	2	1.74	→2.95	2.3	1.28

These data, however, had to be converted to as-generated concentrations for the two compressible waste streams. Information received from a utility official indicates that the compressible wastes were compacted into 98 ft^3 boxes. Total packaged weight averaged about 4,000 lbs including the box, which weighed about 600 lbs (Ref. 21). These data imply a compacted waste density of about 0.56 g/cm^3, which when compared with an industry average for as-generated compressible waste of 0.13 g/cm^3 (Ref. 10) imply an average VRF for this particular utility of 4.27. This VRF was used to convert the as-shipped compressible concentrations to as-generated concentrations. An average VRF equal to unity was assumed for noncompressible waste streams.

A brief review of the manner in which the concentration distributions are used in this report is presented in Section A.9.1.

Table A-12. Distribution of Gross Concentration in LWR Process Waste Streams

Range		P-IXRESIN		P-CONCLIQ		P-FSLUDGE		B-IXRESIN		B-CONCLIQ		B-FSLUDGE	
Ci/ft^3	Ci/m^3	Vol%	Conc	Vol%	Conc	Vol%	Conc	Vol%	Conc	Vol%	Conc	Vol%	Conc
.000005 - .00001	.00018 - .00035							0.1	.00025			0.1	.00025
.00001 - .00005	.00035 - .0018	0.7	.0011	0.4	.0017	0.7	.0011	0	-			0	-
.00005 - .0001	.0018 - .0035	4.8	.0023	0.6	.0024	4.8	.0023	0.4	.0021			0.4	.0021
.0001 - .0005	.0035 - .018	7.8	.0088	4.3	.012	7.8	.0088	3.7	.0096	7.5	.0095	3.7	.0096
.0005 - .001	.018 - .035	6.1	.025	9.1	.024	6.1	.025	2.9	.025	0	-	2.9	.025
.001 - .005	.035 - .18	19.3	.088	28.6	.099	19.3	.088	11.4	.10	45.7	.10	11.4	.10
.005 - .01	.18 - .35	8.3	.26	20.4	.27	8.3	.26	7.2	.26	22.3	.26	7.2	.26
.01 - .05	.35 - 1.8	22.1	.85	30.4	.84	22.1	.85	30.6	.86	8.8	.96	30.6	.86
.05 - .1	1.8 - 3.5	8.5	2.61	3.2	2.37	8.5	2.61	11.8	2.65	6.1	2.63	11.8	2.65
.1 - .5	3.5 - 18	14.3	7.47	2.6	5.65	14.3	7.47	25.7	7.38	9.7	5.33	25.7	7.38
.5 - 1	18 - 35	3.9	25.02	0.1	17.97	3.9	25.02	3.6	25.67			3.6	25.67
1 - 5	35 - 180	3.2	73.28	0.3	86.55	3.2	73.28	2.4	60.63			2.4	60.63
5 - 10	180 - 350	0.8	260.34			0.8	260.34	0.1	253.17			0.1	253.17
10+	350+	0.2	395.17			0.2	395.17	0.1	461.96			0.1	461.96
Weighted average concentration (Ci/m^3):			7.71		0.84		7.71		5.60		0.866		5.60

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Table A-13. Distribution of Gross Concentration in LWR Trash Waste Streams

Concentration Range (Ci/m³)	Volume Percent	Average Concentration in Range (Ci/m³)
P-COTRASH:		
0 - 9.61E-3	26	6.92E-3
9.61E-3 - 1.93E-2	23	1.56E-2
1.93E-2 - 7.70E-2	35	6.22E-2
7.70E-2 - 1.93E-1	12	1.27E-1
> 1.93E-1	4	3.16E-1
	Weighted average:	5.50E-2
P-NCTRASH:		
0 - 4.11E-2	49	2.08E-2
4.11E-2 - 8.22E-2	23	7.16E-2
8.22E-2 - 3.29E-1	20	2.38E-1
3.29E-1 - 8.22E-1	-	-
>8.22E-1	8	3.79E+0
	Weighted average:	3.77E-1
B-COTRASH:		
0 - 6.83E-3	70	4.23E-3
6.83E-3 - 1.37E-2	13	1.22E-2
1.37E-2 - 5.45E-2	11	2.95E-2
5.45E-2 - 1.37E-1	4	9.48E-2
>1.37E-1	2	4.07E-1
	Weighted average:	1.97E-2
B-NCTRASH:		
0 - 2.95E-2	69	1.76E-2
2.95E-2 - 5.90E-2	9	5.27E-2
5.90E-2 - 2.36E-1	15	1.30E-1
2.36E-1 - 5.90E-1	4.7	4.15E-1
>5.90E-1	2.3	1.28E+0
	Weighted average:	8.53E-1

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A.2.2.3 Other Light Water Reactor Waste Streams

Two waste streams are considered here: activated nonfuel reactor core components (L-NFRCOMP) and waste from periodic reactor decontamination (L-DECONRS).

Projected volumes of activated non-fuel core components (e.g., poison curtains, flow channels, control rods, instrumentation) are difficult to characterize. LWR core components are replaced on an infrequent basis, and frequently, small components are shipped to disposal facilities by placing the components in the middle of a container of otherwise low activity material such as trash. For the draft Part 61 EIS, insufficient data existed to distinguish between irradiated components from PWRs and BWRs. A gross LWR generation rate was assumed amounting to 0.001 m^3 of waste per MW(e)-yr at a gross specific activity of about 113 Ci/m^3 (4000 Ci/ft^3). This projection was based upon a limited review of disposal facility radioactive shipment records. More recent data obtained from LWR semi-annual release reports for 1978 through 1982 have been summarized in Table A-10.

Based upon the data in Table A-10, the following generation rate and gross activity are assumed, incorporating an average across both LWR types:

Generation Rate (m^3/MW(e)-yr)	Activity (Ci/MW(e)-yr)	Concentration (Ci/m^3)
2.79E-2	2.67	95.6

The information in the semi-annual release reports is a bit spotty, but sufficient data is judged to be available (barely) to warrant presenting the concentrations in this waste stream as a distribution. The concentration distribution covers 9 orders of magnitude and is given in Table A-14. As shown, most of the activity is either in a range of 0.001 to 0.1 Ci/m^3, or in a range of 10 to 100 Ci/m^3.

Table A-14. Concentration Distribution for L-NFRCOMP Waste Stream

Range (Ci/m^3)	Volume Percent in Range*	Average Concentration in Range (Ci/m^3)
0.0001 $\leq$ 0.001	2.87	5.60E-4
0.001 $\leq$ 0.01	25.5	5.34E-3
0.01 $\leq$ 0.1	17.6	3.93E-2
0.1 $\leq$ 1.0	0.36	2.35E-1
1.0 $\leq$ 10.	2.26	6.54E+0
10. $\leq$ 100.	44.9	1.17E+1
100. $\leq$ 1,000.	4.42	2.11E+2
1,000 $\leq$ 10,000.	1.91	2.77E+3
10,000 $\leq$ 100,000.	0.11	2.53E+4

*Note: Total of 49 data points used.

Other waste streams that are difficult to project will be generated by periodic decontamination of LWR primary coolant systems. Decontamination operations are performed to reduce plant personnel exposures by removing crud accumulated on surfaces in contact with the primary coolant.

There may be a number of possible decontamination processes utilized--e.g., from dilute chemical processes on an annual basis to more concentrated processes at intervals of several years--and the waste streams generated may vary in kind (e.g., resins, solidified liquids) and in volume from operation to operation and plant to plant. Other plant-specific factors which would influence the volumes, radioactivity content, and other characteristics of the wastes generated would include the operating history of the plant (e.g., history of fuel failures), the design of the plant and liquid clean-up and processing systems, the chemistry of the primary coolant, and the length of time between decontamination operations. Institutional matters such as the policies of a specific utility could also be a consideration.

For the draft and final Part 61 EIS, it was assumed that every operating PWR undergoes a full-scale primary coolant decontamination operation every 5 to 10 years using a dilute chemical decontamination operation. This resulted in assumed resin waste streams of about 95 m^3 per operation for a BWR and 47.5 m^3 per operation for a PWR. At the time these projections were made, a full scale decontamination of the Dresden, Unit 1, facility was expected to start shortly (Ref. 22). Considerable follow-up interest by other utilities was expected (Refs. 3,4).

The volumes and activities thus projected were highly speculative, but amounted to a large fraction of the waste projected in the draft and final Part 61 EIS to be disposed by the year 2000. This was believed to potentially result in skewed projections regarding wastes that might exceed Class C concentrations. A review of the waste projections was thus performed for this work.

Since the original projections were made, considerable practical experience in large-scale decontamination activities has been gained. To date, the only full-scale decontamination of primary coolant systems that has been performed has been at Dresden, Unit 1. Decontamination operations were significantly delayed from the originally scheduled date, and the unit is furthermore scheduled to be mothballed. However, a number of less-extensive decontamination operations have been performed at several power plants. Decontamination campaigns through the end of 1984 are summarized on Table A-15 (Refs. 23-31). Total annual as-shipped waste volumes as obtained from Table A-15 are estimated below, assuming a volume increase factor of 1.4 for solidified resins:

Annual As-Shipped Decontamination Waste Volumes (m^3)

Year	BWR	PWR	Total
1976	7.1		7.1
1977	15.3		15.3
1978			
1979	0.7		0.7
1980	0.7		0.7
1981	2.5		2.5
1982	8.8	0.7	9.5
1983	12.3	2.9	15.2
1984	169.*	2.8	172.

*71 m^3 if Dresden 1 is excluded.

Clearly, decontamination is more popular with operators of BWRs than with PWRs. Comparison of these as-shipped waste volumes with the projected country-wide electrical generating capacity (Tables A-8 and A-9) reveals the following general trends:

	Shipped Volumes (m^3/1000 MW(e))	
Year	BWR	PWR
1980	3.8E-2	0
1981	1.4E-1	0
1982	4.6E-1	1.8E-2
1983	5.5E-1	7.3E-2
1984	5.6E+0*	5.2E-2

*3.1 m^3/GW(e) if Dresden 1 is excluded.

It may also be noted that almost 60% of the 1984 BWR volume is involved with the Dresden 1 primary system decontamination, which still involved significantly lower waste volumes than those projected earlier (Ref. 22). These wastes were generated in 1984 but will actually not be solidified and disposed until 1985.

Typical systems decontaminated at BWRs include reactor water cleanup systems and recirculation piping. Experience at PWRs has principally involved decontamination of steam generators. To date, the London Nuclear Candecon system appears to have been the most popular, although a number of other processes have also been marketed within recent years, including the NS-1, Lomi, and Citrox processes. The Lomi and NS-1 processes are planned for use at a PWR and BWR, respectively, in 1985. Two other BWRs will also be decontaminated in 1985 using the Lomi process at one and either the Lomi or the Citrox process in another. Steam generators for at least two additional PWRs will be decontaminated, one using the new EPRI process. This last operation will involve extensive decontamination of a pair of steam generators, and is expected to generate up to approximately 180 m^3 of dewatered resins (15% of volume) and solidified liquids (Refs. 23-31). A description of a variety of decontamination processes is contained in reference 32.

Table A-15. Summary of LWR Decontamination Experience

Year	Plant	Type	NRC Region	System	Process	Waste Vol. (m^3)	Removed Activity (Ci)	Conc. (Ci/m^3)	Waste Form
1976	Dresden 1	B	III	Test loop	Dow NS-1	7.1(a)	~20	2.8(a)	VES solid
1977	Peach Bottom	B	I	Regen. heat exchangers	Dow NS-1	15.3(a)	17	1.1(a)	VES solid
1979	Vermont Yankee	B	I	RWCS	LN Candecon	0.7	2.4	3.4	Dewatered resins
1980	Brunswick 2	B	II	RWCS	LN Candecon	0.7	9.1	13.0	Dewatered resins
1981	Nine Mile Point	B	I	Recirc. pumps	LN Candecon	1.1	38.7	35.2	Dewatered resins
1981	Brunswick 1	B	II	RWCS	LN Candecon	0.7	17.0	24.3	Dewatered resins
1981	Vermont Yankee	B	I	RWCS	LN Candecon	0.7	5.0	7.1	Dewatered resins
1982	Peach Bottom 2	B	I	RWCS	LN Candecon	0.7	31.3	44.7	Dewatered resins
1982	Nine Mile Point	B	I	Recirc. system	LN Candecon	2.5	67.6	27.0	Dewatered resins in HIC
1982	Hatch 1	B	II	RWCS	LN Candecon	1.4	2.3	1.6	Dewatered resins
1982	Brunswick	B	II	Aux. steam & cond.	LN Candecon	4.2	0.1	0.02	Dewatered resins & filter cartridge
1982	Surry 2 at Battelle	P	II	Steam generator	LN Candecon	0.7	2.1	3.0	Dewatered resins
1983	Peach Bottom 3	B	I	RWCS	LN Candecon	2.7	36.7	13.6	Dewatered resins
1983	Dresden 3	B	III	RWCS	LN Candecon	0.4(b)	20.6	51.5(b)	Cement solid. resins
1983	Vermont Yankee	B	I	Recirc. System	LN Candecon	5.4	264.7	49.0	Resins
1983	Quad Cities 2	B	III	Recirc. System	LN Candecon	1.3(b)	153.8	118.3(b)	Cement solid. resins
1983	Dresden 3	B	III	Recirc. System	LN Candecon	1.3(b)	108.3	83.3(b)	Cement solid. resins
1983	Ginna	P	I	Steam generator	LN Candecon	2.1(b)	71.2	33.9(b)	Cement solid. resins
1984	Quad Cities 2	B	III	RWCS	LN Candecon	0.14	2.0	14.3	Resins
1984	Quad Cities 1	B	III	RWCS	LN Candecon	0.4(b)	12.2	30.5(b)	Cement solid. resins
1984	Millstone 1	B	I	RWCS	LN Candecon	0.4	4.4	11.0	Resins
1984	Quad Cities 1	B	III	Recirc. System	LN Candecon	1.1(b)	117.8	107.1(b)	Cement solid. resins
1984	Millstone 1	B	I	Recirc. System	LN Candecon	1.3(b)	54.6	42.0(b)	Cement solid. resins
1984	Peach Bottom 2	B	I	Recirc. System	LN Candecon	1.4	46.0	32.9	Resins

Table A-15. (Continued)

Year	Plant	Type	NRC Region	System	Process	Waste Vol. (m^3)	Removed Activity (Ci)	Conc. (Ci/m^3)	Waste Form
1984	Palisades	P	III	Steam generator	LN Candecon	2.0(b)	15.4	7.7(b)	Cement solid. resins
1984	Dresden 1	B	III	Primary System	IT NS-1	~70(b)	~800	~11.4(b)	Resins (14%) and liq. to be cement solid. in 1985
1984	Pilgrim	B	I	Recirc. System	IT NS-1	9.3(a)	79.4	8.5(a)	Cement solid. liq. and resins (27%)
1984	Pilgrim	B	I	RWCS	IT/PN NS-1	5.3(b)	4.2	0.8(b)	Cement solid. liq., resins, and filter cartridge
1984	Dresden 2	B	III	Recirc. & RWCS	IT/PN NS-1	7.4(b)	72	9.7(b)	Cement solid. resins
1984	Monticello	B	III	Recirc. System	Quadrex Lomi	19.8(b)	59.1	3.0(b)	Cement solid. resins + filter cartridges (.4m^3)
1984	Brunswick 2	B	II	RWCS	PN Citrox	1.9(b)	10-15	~6.6(b)	Cement solid. resins + filter cartridges (.2m^3)
1984	Cooper	B	IV	Recirc. System	PN Citrox	5.2(b)	19.4	3.8(b)	Cement solid. resins + filter cartridges (.2m^3)

Abbreviations: VES: vinyl ester styrene; RWCS: reactor water cleanup system; B: BWR; P: Pwr; LN: London Nuclear; IT: IT Corporation; PN: Pacific Nuclear; HIC: high integrity container.

(a) Volumes and concentrations for solidified waste form.
(b) Volumes and concentrations for unsolidified waste form.
(c) Volume does not include filter.

Future waste volume projections are difficult to make, although it seems reasonable to expect that activities in at least the near future will parallel those in the near past. Full scale primary coolant system decontamination, such as that performed at Dresden 1, will probably continue to be performed on an infrequent basis. However, a number of significantly less extensive decontamination campaigns will continue to be carried out closely tied to specific maintenance requirements. BWRs will probably continue to be the most common customers of solidification services. PWRs will use the services to some extent, mainly involving maintenance of steam generators, although perhaps not tied as closely to tube denting problems. Service vendors will continue to improve the processes to achieve higher decontamination factors and smaller waste volumes.

For this report, some very approximate assumptions are made for the L-DECONRS waste stream. First, it is assumed that the waste stream can be characterized as ion-exchange resins containing significant quantities of chelating agents. Second, volume generation rates are assumed to be proportional to installed electrical generating capacity. As-generated (dewatered) volumes are assumed to be roughly 3 m^3/GW(e) for BWRs and 0.1 m^3/GW(e) for PWRs. This is based on the assumption that extensive decontamination operations involving primary systems are performed infrequently. The regional generation rate is assumed to follow the regional electrical capacity.

In terms of radionuclide concentrations, a review of the data in Table A-15 indicates the following concentration distribution, where the volumes and concentrations are given an approximate as-generated condition:

Concentration Range (Ci/m^3)	Volume (m^3)	Volume Percent
$<.1$	4.2	2.6
$.1 \leq .5$		
$.5 \leq 1.$	5.3	3.3
$1. \leq 5.$	35.5	22.1
$5. \leq 10.$	22.2	13.8
$10. \leq 50.$	89.5	55.7
$50. \leq 100.$	1.7	1.1
$\geq 100.$	2.4	1.5

As shown, the spread in the concentration distribution does not appear to be especially large. About 70% of the 32 data points appears to be in a range of 5 to 50 Ci/m^3, while about 92% appears to be in a range of 1 to 50 Ci/m^3. Given this, and also given the current small data base, an average radionuclide concentration is assumed across the waste stream. This is taken to be roughly 25 Ci/m^3. Continued observation and review of decontamination activities at light water reactors is warranted to improve the data base.

A.3 OTHER NUCLEAR FUEL CYCLE FACILITIES

Other nuclear fuel cycle waste streams considered in this report include process wastes from uranium hexafluoride (UF_6) conversion plants and fuel fabrication facilities, trash from fuel fabrication facilities, and waste from decontamination and operation of existing fuel research facilities. These wastes are generally not well characterized.

A.3.1 Waste Generation Overview

Processed uranium ore or yellowcake contains about 0.7 percent fissile U-235 and must be enriched in U-235 content prior to utilization in LWRs. Prior to enrichment by the gaseous diffusion process (the major technology currently used for enrichment), the uranium must be converted to UF_6, which is an easily volatilized compound suitable for this process. During this process in UF_6 conversion facilities, liquid and solid wastes are generated. Many of these waste streams are recycled in the plant to recover uranium. Some process wastes, however, are shipped for disposal. These wastes consist primarily of calcium fluoride generated in hydrogen fluoride gas scrubbers, bed materials from fluidized bed reactors, and lime from treatment of liquid effluents.

Fuel fabrication is the final step before uranium fuel is utilized in LWRs. In fuel fabrication facilities, enriched UF_6 from gaseous diffusion plants is converted into a solid form (usually uranium dioxide) and then into fuel pellets, fuel rods, and finally fuel assemblies. A large portion of the wastes generated during these operations is recycled to recover uranium. Process wastes shipped for disposal include limestone used in calcium fluoride scrubbers, calcium fluoride, oxides from calciners, filter sludges, and small amounts of oils. Trash shipped for disposal includes paper, plastic, equipment, and miscellaneous combustible materials.

Waste from decontamination and operation of existing plutonium fuel research has been discussed in reference 4. At one time several licensees operated small facilities for research and development of plutonium and mixed oxide fuels. These facilities produced fuel rods for experimental use in test reactors and occasionally in commercial nuclear power plants (that is, occasionally one or two fuel rods in a replacement fuel assembly would be loaded with MOX fuel rather than the normal UO_2 fuel). Some licensees would also operate hot cells in which fuel rods removed from fuel assemblies would be subjected to radiochemical analysis to verify fuel burnup calculations and the radiochemical composition of the spent fuel.

The policy of deferring reprocessing has halted most of the mixed oxide fuel research and development in the commercial sector. (Some research is still being carried on, however, either by or under contract to DOE.) Most commercial mixed oxide fuel fabrication test facilities implemented a program for facility clean-up and decontamination. Waste will thus be produced during these decontamination operations, and also occasionally as part of fuel burnup studies.

The radiological, physical, and chemical characteristics of the waste so generated are expected to be quite varied. Waste generated from decontamination operations is expected to be a mixture of compressible and noncompressible material, including such material as paper, miscellaneous trash, glove boxes, and concrete rubble. Such waste would be expected to contain at least some quantities of chelating and other decontamination chemicals. Waste from burnup studies would be expected to mainly consist of compressible trash, with smaller traces of oxide residues.

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A.3.2 Waste Stream Projections

A.3.2.1 Uranium Conversion and Fuel Fabrication Waste

Fuel fabrication wastes include process wastes, compactible (combustible) trash, and noncompactible (noncombustible) trash. These waste streams account for only a small fraction of the waste volume sent to disposal facilities and are also radiologically insignificant by comparison. Given this, no change is assumed for the projections for these waste streams given in reference 1. The waste volumes are again given in terms of electrical generating and capacity and are listed below along with the assumed average gross concentrations:

Waste Stream		Generation Rate	Ave. Gross Concentration
Symbol	Description	(m^3/MW(e)-yr)	(Ci/m^3)
F-PROCESS	process waste	2.68E-2	1.08E-4
F-COTRASH	compactible trash	8.09E-2	5.58E-6
F-NCTRASH	noncompactible trash	1.43E-2	5.33E-6

Volume and concentration projections for the F-PROCESS and F-NCTRASH waste streams are given as ready for shipment, while volume and concentration projections for the F-COTRASH waste stream are given as generated. Further waste processing such as compaction would lower the volume and raise the concentration. The regional distribution is again given as follows: NRC Region I: 20%, NRC Region II: 50%, and NRC Region V: 30%.

Projections for uranium conversion wastes (U-PROCESS) are also adopted without change from reference 1. These projections are given as ready for shipment and are as follows: 9.64E-3 m^3 per MW(e)-yr, and 3.81E-4 Ci/m^3. The two existing commercial UF_6 conversion facilities are located at Metropolis, Illinois and Sequoyah, Oklahoma. As a result, 50% of the waste is assumed to be generated in NRC Region III while the remainder is assumed to be generated in NRC Region IV.

A.3.2.2 Plutonium Facility Decontamination and Fuel Burnup Studies

As discussed in the final Part 61 EIS (Ref. 4), there are only small quantities of waste contaminated with transuranic (TRU) isotopes currently being generated by the commercial sector. Future sources of TRU waste may include (1) recycle of spent uranium fuel, (2) sealed source devices using transuranic isotopes (e.g., old types of neutron generators, americium-241 well logging sources, Pu-238 batteries), (3) waste from manufacture of sealed sources containing transuranic isotopes, (4) decontamination of existing small plutonium research and fuel fabrication facilities, and (5) burnup studies of irradiated LWR fuel. For a number of reasons which have been discussed in Appendix D of reference 4, the authors do not expect significant quantities of waste from commercial fuel recycle activities--at least not until after the year 2000 if at all. For perspective, however, estimates of waste that would be generated by such recycle activities have been included in Section A.4.7.1 of this report. Waste sealed source devices and associated manufacturing waste are considered in Section A.4.5. The remainder of this section addresses the latter two sources of waste.

At one time, there was considerable interest in the energy option of reprocessing spent reactor fuel, followed by manufacture of the recovered plutonium into new fuel rods which would be used by light water reactors. Rather than the current arrangement in which all of the fuel loaded into light water reactors would consist of an enriched uranium dioxide, some or all of the fuel would be a mixture of plutonium and uranium oxides. A number of facilities had programs to research and develop such mixed oxide (MOX) fuels, including fabrication of small quantities of MOX fuel rods for test purposes in light water and other types of reactors, and performing radiochemical analysis of irradiated MOX fuel to acquire burnup and other data. However, the only commercial reprocessing facility ever to operate in the United States has been closed since 1972 (the West Valley, New York facility). In addition, President Carter in 1977 announced a national policy of deferment of commercial fuel reprocessing. This policy of deferring fuel reprocessing halted most of the mixed oxide fuel research and development work in the commercial sector. (Some research is still being carried on, however, either by or under contract to DOE.) Operators of the commercial mixed oxide fuel fabrication test facilities implemented programs for facility cleanup and decontamination.

In addition, occasional radiochemical analyses are carried out on irradiated light water reactor fuel rods. These analyses are carried out independent of plutonium fuel studies, and are intended to help obtain data such as burnup, fission spectra, and so forth. The analyses generally involve chopping one or two selected fuel rods into small segments and leaching out the exposed irradiated fuel. The dissolved fuel is then analyzed. Radioactive waste containing or contaminated with transuranic and other isotopes is generated as a result.

A factor influencing the generation of TRU waste has been a lack of capacity for TRU waste disposal. Individual disposal facility license conditions have imposed a 10 nanocurie per gram (10 nCi/gm) disposal limit for TRU waste at all operating commercial low-level waste disposal facilities. Although at one time five of the six commercial LLW disposal sites accepted TRU wastes for disposal in essentially unlimited concentrations (the Barnwell, South Carolina facility has never accepted TRU wastes for disposal), this practice was gradually discontinued. The last commercial facility to accept TRU waste for disposal was the facility located in the center of the Hanford Reservation near Richland, Washington and operated by the Nuclear Engineering Company (NECO, now U.S. Ecology, Inc.). From 1976 to 1979, this facility was the only commercial disposal facility accepting TRU waste for disposal. TRU waste acceptance at this facility in concentrations exceeding 10 nCi/gm was prohibited by the State of Washington in November 1979. This prohibition hampered decontamination activities at the MOX fuel research facilities discussed above.

Since the effective date of the waste classification portion of the Part 61 regulation (December 27, 1983), there has been some relief from this restriction. According to the regulation, waste not exceeding TRU concentrations of 10 nCi/gm may be disposed as Class A waste while waste containing certain TRU isotopes in concentrations about 10 nCi/gm but not exceeding 100 nCi/gm may be disposed as Class C waste. License conditions at some of the existing operating disposal facilities have since been modified to allow, under facility-specific conditions, disposal of transuranic contaminated waste not exceeding 100 nCi/gm. In addition, some of the decontamination waste generated at the MOX fuel research facilities is being transferred to DOE for storage. This has been brought about due to past contractual arrangements with either DOE or its predecessors. (That is, past government contracts resulted in much of the contamination being currently removed.)

Thus, two waste systems are considered:

Designation	Description
L-PUDECON	Waste from decontamination of MOX fuel fabrication facilities
L-BURNUPS	Waste from operation of hot cells conducting fuel burnup studies

L-PUDECON

This waste stream is expected to consist of a mixture of compressible and noncompressible material, including such material as paper, rubber gloves, plexiglass, metal, plastic, contaminated clothing, and other miscellaneous trash. Such waste would be expected to contain quantities of chelating and other decontamination chemicals. It will be principally generated at two facilities.

One facility located in NRC Region V is currently undergoing decontamination operations which will be completed in the fall of 1985. At this time, approximately 225 ft^3 (10.6 m^3) of waste will have been generated which is expected to exceed Class C concentrations. This waste will be safely stored until offsite storage or disposal capacity is available. Waste generation at a facility located in NRC Region II is expected to be much larger in volume, and extensive decontamination activities are therefore not expected until offsite storage or disposal capacity is available. This would most likely entail transfer to DOE for storage until suitable disposal arrangements are made. Assuming that offsite capacity is available, it is estimated that roughly 6-9 months would be required to prepare for decontamination operations, including hiring additional staff. After that, actual decontaminational operations will require about 3 years, resulting in about 5,000 to 10,000 ft^3 of waste which is projected to have transuranic concentrations exceeding 100 nCi/g. Some waste will also be generated which is less than 100 nCi/g in transuranic concentration, of which a significant fraction should be less than 10 nCi/g (Refs. 33-35).

This projection will only concern itself with waste having transuranic concentrations greater than 100 nCi/gm. This waste is assumed to be generated in NRC Regions II and V as follows:

L-PUDECON

Region	Volume (m^3)
II	212
V	10.6

For this report, waste from the facility in NRC Region V is assumed to be generated in only a single year: 1985. Waste from the facility in NRC Region II is assumed to be generated starting in 1987. This waste will be generated for only three years at a rate of 70.7 m^3 per year.

L-BURNUPS

At present, three facilities perform post-irradiation radiochemical inspection of nuclear fuels, during which small quantities of very high activity waste are generated. These facilities (hot cells) are located, respectively, in NRC Regions II, III, and V. A past bounding estimate was that each generates about 200 ft^3 of waste per year (Ref. 35), although more recent data indicates that this projection is significantly excessive. A description of operations and waste generation at one facility follows (Ref. 36).

> Fuel elements are sectioned in a hot cell, using a diamond wheel cutter. Sections through a UO_2 fuel pellet are taken for metallographic examination and a sample roughly 1/2 inch in length is taken for burnup analysis. The Zircaloy cladding is removed mechanically. The sections for metallographic examination are transferred to a metallurgy cell for polishing. The sample for burnup analysis is dissolved in the hot cell to a final volume of 100 mL. A small aliquot of this solution, such as 0.1 mL, is diluted by a factor of 10^4 to make up the solution which is used for the analytical work.
>
> The "hot" liquid solutions produced by dissolving the samples of fuel are estimated to comprise roughly 2/3 of the TRU activity in the waste and this is contained in 5-6% of the volume. The dilute solutions on which the burnup analyses are performed contain only ~0.1% of the activity in the hot liquid waste stream, but make up roughly 2/3 of the total volume. Both types of liquid waste solution (acidic when prepared) are neutralized and then solidified with cement in a volume ratio of 2:1 cement to liquid, in either pint or gallon cans.
>
> The remainder of the waste, approximately one-third of the activity and 1/4 of the volume, consists of solid waste, which is placed in both pint and gallon cans, without encapsulation in cement. The waste contains the cuttings from the fuel sectioning (picked up on wipe papers), the cutting wheels, pieces of Zircaloy cladding, the metallographic polished sections in their plastic mounts, contaminated glassware and other contaminated equipment. Another contribution to this waste comes from the polishing operations. It is made up of particles of fuel and polishing compound which collect in a sump. The sump is cleaned out occasionally and the sludge is solidified in cement. The total volume of this waste is quite small, and the TRU level is estimated to be considerably lower than that in the solidified hot liquid waste.

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A summary of the waste generated during these operations is provided below (Ref. 36).

Waste Stream	Stored Volume (m^3)*	Estimated Annual Production (m^3/yr)	Estimated TRU Activity (Ci)	Average TRU Content (nCi/g)
Solidified hot liquid from dissolving fuel samples	0.09	0.01	135	1.5E+6
Solidified dilute solutions from burnup analysis	0.96	0.11	0.01	80-100
Solid waste	0.43	0.05	72	6.0E+5
	1.48	0.17	207	

*As of April 1984.

This waste is currently all being stored. Other waste from hot cell operations, which includes non-transuranic waste from examination work on reactor fuel and components, fabrication of Co-60 and Cf-252 sources, and the production of radiopharmaceuticals, is packaged and shipped to low-level waste disposal sites. This waste is considerably larger in volume.

For this report, a total annual transuranic waste volume of 200 ft^3 (5.7 m^3) is conservatively assumed to be generated by the three facilities (Refs. 36, 37). The waste is assumed to be distributed as follows (in m^3):

Year	NRC Reg. II	NRC Reg. III	NRC Reg. V
1984*	8.1	11.3	1.65
1985+**	2.5	3.0	0.2

*Represents waste stored through the end of 1984 (m^3).

**Represents annual addition (m^3/yr).

A.4 INSTITUTIONAL FACILITIES

Institutional waste generators include colleges and universities, medical schools, research facilities, and hospitals. These institutions use radioactive materials in many diverse applications. Sealed sources and foils are widely used as integral parts of analytical instruments and irradiators. Labelled pharmaceuticals and biochemicals are used in nuclear medicine for therapy and diagnosis, and in biological research to study the physiology of humans, animals, and plants. Radioactive materials are also used by many other academic disciplines such as chemistry, physics, and engineering. Radioactive waste streams are also produced by institutions as a byproduct of research using neutron activation analysis, particle accelerators, and research reactors.

A.4.1 Waste Generation Overview

Based upon information received from surveys (Refs. 38, 39), institutional wastes may be classified into four volumetrically significant groups: liquid scintillation vials containing scintillation fluid (shipped with absorbent materials), other liquids (solidified or shipped with absorbent materials), biological wastes (shipped with absorbent materials and lime), and trash. In addition to these streams, institutional facilities occasionally generate two volumetrically smaller waste streams, accelerator targets and sealed sources, that have been included under the next section on industrial wastes.

Liquid scintillation counting techniques are used to some extent by nearly all varieties of fuel cycle and nonfuel cycle waste generators; however, applications in biological research produce the major volumes of waste scintillation vials and fluids. The vials are made of glass and occasionally polyethylene, and are usually about half full of counting fluid. Flammable organic solvents (e.g., toluene, benzene, xylene) comprise the major consitituents of scintillation fluids.

Absorbed liquids have not been as well characterized as liquid scintillation vials, in part because the composition of absorbed liquids is not constrained by the requirements of liquid scintillation counting techniques. Approximately 50% of these absorbed liquids are scintillation fluids. The remaining liquids are aqueous and organic solvents generated by diverse preparatory and analytical procedures such as wastes from elution of Tc-99m generators, radioimmunoassay procedures, and tracer studies.

Biological wastes are generated primarily through research programs at universities and at medical schools. The waste consists of animal carcasses, tissues, animal bedding, and excreta, as well as vegetation and culture media. Radioactive excreta from humans undergoing diagnostic or therapeutic procedures that use radioactive materials are not included since virtually all such materials are discharged to sewers.

Institutional trash consists almost entirely of materials that are both compactible and combustible. It generally consists of paper, rubber or plastic gloves, disposable and broken labware, and disposable syringes.

A.4.2 Waste Stream Projections

Institutional wastes are generally (but not always) low in activity. Projections made for these wastes in reference 1 identified four waste streams. These included dry solids and other trash, scintillation liquids and vials, other liquids (mostly aqueous), and animal carcasses and other biological wastes. Volumes were assumed to increase at a linear rate each year, and a gross average concentration was assumed for each type of waste. The assumed as-generated volumes for each type of waste were as follows (Ref. 1):

Type of waste	Volumes (m^3) In 1980	Volumes (m^3) Added per year
Dry solids	8,028	538
Scintillation liquids and vials	2,804	188
Other liquids	318	21
Biological	996	60

The above volumes are for "untreated" waste--i.e., waste prior to packaging for shipment to a disposal facility. Packaging considerations will alter the projected volumes. For example, transportation considerations and disposal facility license conditions require that liquids, scintillation vials, and biological wastes be packaged according to certain prescribed requirements, including the use of absorbents, and the resulting packaged volume of the waste will be significantly larger than the untreated volumes. This is assumed to be about a factor of 3 for scintillation vials and scintillation and other liquids, and a factor slightly less than 2 for biological wastes.

Since these projections were made, NRC promulgated a new Section 20.306 to 10 CFR Part 20 allowing disposal of biological waste and scintillation liquids and vials by less restrictive means, provided that such waste contained tritium or carbon-14 in concentrations less than 0.05 μCi/gm. This new regulation resulted in a significant reduction in the amount of scintillation and biological waste being shipped to low-level waste disposal sites. Disposal records and disposal site operators were consulted to determine the volumes of actual waste shipments to disposal sites. In 1980, it was determined that shipments of scintillation waste to disposal sites totaled about 199,000 ft^3. For 1982, the total scintillation waste volume (absorbed scintillation liquid, solidified scintillation liquid, and scintillation material in vials) shipped to disposal sites was determined to be about 101,000 ft^3; for 1983, this total came to about 80,000 ft^3. Biological waste shipments totaled 25,600 ft^3 in 1982 and 25,500 ft^3 in 1983 (Ref. 40). These volumes include approximate volume increase factors during packaging of about 3 for scintillation waste and about 2 for biological waste.

Data from 1984 indicates further reductions in the quantity of scintillation waste being shipped to low-level waste disposal sites. Total national disposed volumes are listed below (Ref. 83).

Scintillation Volumes Disposed in 1984 (ft^3)

	Regulated	Deregulated	Total
Combined	17,599.28	6,040.07	23,639.35
Organic	10,039.46	3,664.79	13,704.25
Aqueous	2,030.23	390.76	2,420.99
Total	29,668.97	8,461.06	39,764.59

In the above table, the "organic" heading refers to waste which contains organic scintillation counting media such as toluene or xylene, while the "aqueous" heading refers to waste which contains nonorganic scintillation counting media. The "combined" heading refers to waste which may or may not contain organic material. Finally, the "regulated" heading refers to waste which exceeds the concentration limits in Section 20.306 of 10 CFR Part 20, while the "deregulated" heading refers to waste that has concentrations less than the limits. Whatever the concentrations, however, all the listed waste was disposed at licensed low-level waste disposal sites. Waste disposed by other means pursuant to Section 20.306 is not included.

Based on the above, the following as-generated projections are made for the biological and scintillation waste streams:

Year	As-Generated Volume (m^3)	
	Scintillation Waste	Biological Waste
1980	1876	996
1981	1416	687
1982	956	378
1983	752	376
1984	375	376

Following 1984, it is conservatively assumed that scintillation and biological waste volumes remain constant. It is believed that any new generation of scintillation and biological wastes will do so having the provisions of Section 20.306 in mind, and will so arrange operations so that waste requiring shipment to a commercial low-level waste disposal site is minimized. In addition, as existing and any new generators expand operations, any new waste generation will probably be countered by further waste reductions by existing generators. Waste generators are making increased use of alternative disposal methods, such as incineration in the case of scintillation media. Generators will also undoubtedly make significantly increased use of nonorganic scintillation media.

Projections of institutional trash and absorbed liquids are assumed to be the same as those given in reference 1. This is believed to be conservative, since the requirements in Section 20.306 also allowed greater use of sanitary sewerage disposal of aqueous liquids containing tritium or carbon-14. This will probably act to reduce the absorbed liquid volumes being sent to disposal sites. No data is in hand, however, to confirm this.

The projections of institutional waste generation in this report also take into account the interest of institutional waste generators in volume reduction techniques. In order to consider the potential for volume reduction, and considering the fact the institutional generators range in size from small to very large operations, the above four waste types are split into the eight specific waste streams considered in this appendix. This is done to identify the volumes of waste from generators that can more easily implement their own waste treatment processes (e.g., comparatively large facilities, denoted by a minus sign), and the waste from those generators which cannot do the same (e.g., comparatively small facilities, denoted by a plus sign). The comparative waste volume produced by each type of waste generator is assumed to be the same. The resulting waste streams, untreated volumes, and gross average (untreated) concentrations are as follows for trash and absorbed liquids:

Waste Stream		Volume (m^3)		Gross Average Concentration
Symbol	Description	In 1980	Added per year*	(Ci/m^3)
I-COTRASH	Trash (large facilities)	4,014	269	1.13E-1
I+COTRASH	Trash (small facilities)	4,014	269	1.13E-1
I-ABSLIQD	Absorbed liquids (large facilities)	159	10.5	1.99E-1
I+ABSLIQD	Absorbed liquids (small facilities)	159	10.5	1.99E-1

*Following the year 2000, generated waste volumes are assumed to be constant for all four waste streams.

For scintillation and biological waste, the resulting waste streams and untreated volumes are as follows:

Waste Stream		Yearly Volume (m^3/yr)				
Symbol	Description	1980	1981	1982	1983	1984+
I-LIQSCVL	Scintillation waste (large facilities)	938	708	478	376	188
I+LIQSCVL	Scintillation waste (small facilities)	938	708	478	376	188
I-BIOWAST	Biowaste (large facilities)	498	343.5	189	188	188
I+BIOWAST	Biowaste (small facilities)	498	343.5	189	188	188

Based on reference 1, average gross concentrations for the two LIQSCVL waste streams are assumed to each be 7.60E-3 Ci/m^3, while the average gross concentrations for the two BIOWAST waste streams are assumed to each be 2.06E-1 Ci/m^3.

The regional distribution of each of the above eight waste streams is assumed to be as follows: NRC Region I: 31%, NRC Region II: 22%, NRC Region III: 27%, NRC Region IV: 8%, and NRC Region V: 12%.

A.5 INDUSTRIAL FACILITIES

Wastes from industrial facilities are difficult to project. At this time, such wastes can be grouped into eight basic groups, two of which appear to be volumetrically significant but of relatively low activity. These include: (1) those from industrial facilities using source and special nuclear materials (generating trash and other miscellaneous wastes), and (2) those from other industrial facilities that use radioactive material and generate low specific activity wastes containing less than 3.5 Ci/m^3 (0.1 Ci/ft^3). Waste from both these groups of facilities is divided into trash and miscellaneous other wastes.

Six groups of waste streams are relatively small in volume but may be relatively high in activity: medical isotope production waste, waste from large industrial manufacturers using tritium and carbon-14, waste from small industrial tritium manufacturers and users, activated metal wastes, sealed sources, and sealed source manufacturing waste.

A.5.1 Low Activity Waste Streams

This group of waste streams includes waste generated as a result of operations involving source and special nuclear material as well as other low activity waste.

Source and special nuclear material wastes are produced outside the nuclear fuel cycle by industrial facilities that process and fabricate depleted uranium and manufacture chemicals or products containing uranium. Although little information is available, it appears that most of the waste is generated through processing of depleted uranium. These wastes are distinguished from other nonfuel cycle wastes by the almost complete absence of radionuclides other than those included in the definitions of source and special nuclear materials. These wastes are considered to be two separate streams: trash (SSTRASH) and other miscellaneous wastes (SSWASTE).

The second group of waste streams consists of low specific activity wastes having concentrations less than 3.5 Ci/m^3 (0.1 Ci/ft^3). The major contributors to this group of streams are the industrial equivalents of institutions. Such waste is generated by pharmaceutical companies, independent testing laboratories, and analytical laboratories. The characteristics of low specific activity industrial wastes are expected to resemble those of institutional wastes; however, since the limited data available is insufficient to justify separate waste streams for scintillation fluids, absorbed liquids, and biowastes, they are also considered as two streams: trash (LOTRASH) and other miscellaneous wastes (LOWASTE).

These two groups of waste streams were characterized in reference 1 and the draft and final Part 61 EIS (Refs. 3, 4). In keeping with the scope of this report, projections of these lower activity industrial waste streams are generally adopted from the draft Part 61 EIS with little change. These are all classified as Class A waste under 10 CFR Part 61. The projected volumes and gross radionuclide concentrations for the above low activity waste streams are as follows:

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Waste Stream		Volume (m^3)		Average Gross Concentration
Symbol	Description	In 1980	Added per year**	(Ci/m^3)
N-SSTRASH	SS* trash (large facilities)	5,122	343	1.12E-5
N+SSTRASH	SS* trash (small facilities)	5,122	343	1.12E-5
N-SSWASTE	other SS waste	1,808	121	2.17E-4
N-LOTRASH	low act. trash (large facilities)	1,445	97	3.53E-2
N+LOTRASH	low act. (small facilities)	1,445	97	3.53E-2
N-LOWASTE	other low activity waste	1,719	115	2.11E-2

*SS: Source and special nuclear material

**Following the year 2000, generated waste volumes are assumed to be constant for all six waste streams.

As before, some waste streams (SSTRASH and LOTRASH) are divided into two waste streams having equal volumes. This is done to distinguish between larger and smaller waste generators. In addition, the volumes for all industrial waste streams except for the four trash streams are given as packaged for shipment. The trash waste streams are given as generated prior to packaging. Compaction and other waste volume reduction techniques would decrease trash volumes during packaging.

The regional distribution for the three source and special nuclear waste streams are assumed to be as follows: NRC Region I: 50%, NRC Region II: 10%, NRC Region III: 20%, NRC Region IV: 10%, NRC Region V: 10%. All other low activity industrial waste streams are assumed to have similar distributions to the institutional waste streams--i.e.: NRC Region I: 30%, NRC Region II: 20%, Region III: 30%, NRC Region IV: 10%, and NRC Region V: 10%.

A.5.2 Higher Activity Waste Streams

The higher activity waste streams consist of medical isotope production waste, waste from large industrial manufacturers using tritium and carbon-14, sealed source manufacturing waste, waste from small tritium manufacturers and users, discarded sealed sources and devices, and activated metals. Considerable improvement in and modifications to the data base have been made since the draft Part 61 EIS.

A.5.2.1 Medical Isotope Production Waste

One of the largest generators of radioisotopes used for medical purposes is located in Tuxedo, New York (NRC Region I). The principal commercial product--Mo-99--is produced as a fission product from neutron irradiation of targets containing highly enriched uranium. A typical target consists of an aluminum secondary capsule (which is removed before processing) enclosing a stainless steel capsule, coated on the inside with UO_2 containing uranium enriched to 93% in the U-235 isotope. The target material is then irradiated in the neutron flux of a low power reactor. Less than 1% of the uranium target material is fissioned during the irradiation (Ref. 41).

Processing is done in a hot cell. Through a fitting in the capsule, a solution of sulfuric acid and nitric acid is injected into the target. The target is heated and rolled on its side to dissolve the UO_2 coating and to free the fission product gases, which are subsequently released and condensed in a copper cold finger immersed in liquid nitrogen. The liquid remaining in the target is removed. In a patented process, 2% alpha-benzoin oxime is added in excess, causing the precipitation of the Mo-99 chelate. The filtrate from the Mo precipitation contains the remaining uranium, the fission products and trace amounts of Pu-239 and other Pu isotopes produced from the neutron activation of U-238. Following additional processing by precipitation, a significant portion of the uranium and about 50% of the fission products are sent to Savannah River Plant (SRP) for reprocessing. The balance of the uranium, fission products, and activation products becomes a waste stream (Ref. 41).

Mo-99, with a half life of 66.6 hours, decays to Tc-99m (6-hour half-life), an isotope which has become an extremely useful tool in medical applications, particularly as a radionuclide tracer in both medical research and applications. Other commercial products, which may be produced either by irradiation of appropriate target materials or by chemical separation from other fission products formed during the above Mo-99 production, include the following (Ref. 41):

1. P-32 as H_3PO_4.
2. Sn-113 as $SnCl_4$ in 4N HCl.
3. I-125 as NaI in NaOH.
4. I-131 as NaI in 0.05N NaOH.
5. Xe-133 as elemental Xe in air.
6. Au-198 as a solute in HCl and HNO_3.

Waste from facility operation can be grouped into two waste streams (N-ISOPROD and N-ISOTRSH). The first waste stream is of high activity and contains various waste products from the hot cell operations, while the second waste stream is of much lower activity and consists of miscellaneous trash and other material generated during balance-of-plant operations.

The high activity waste stream (N-ISOPROD) consists of various materials randomly placed within a 55-gallon waste drum lined with a layer (about 2.5 liters or 25 lbs) of an absorbent, attapulgite clay (See Figure A.4). Such materials can include 304L SS containers holding cement-solidified liquid/sludge fission product waste; neutron-activated 304 SS irradiation targets; copper cold trap tubes; and other miscellaneous materials such as scrap machinery (e.g., vacuum pumps), celluostic pads, compressed paint cans (containing compactible waste such as glass, paper, or plastic), and plastic tubing. A typical waste container weighs about 300 pounds.

Most of the activity is contained in the 304 SS containers containing the solidified liquid/sludge mixture. Such containers are about 5 inches in diameter and 12 inches tall, with a wall thickness of 65 mils. Plates are welded on the bottom and top, with a protruding nozzle welded to the top

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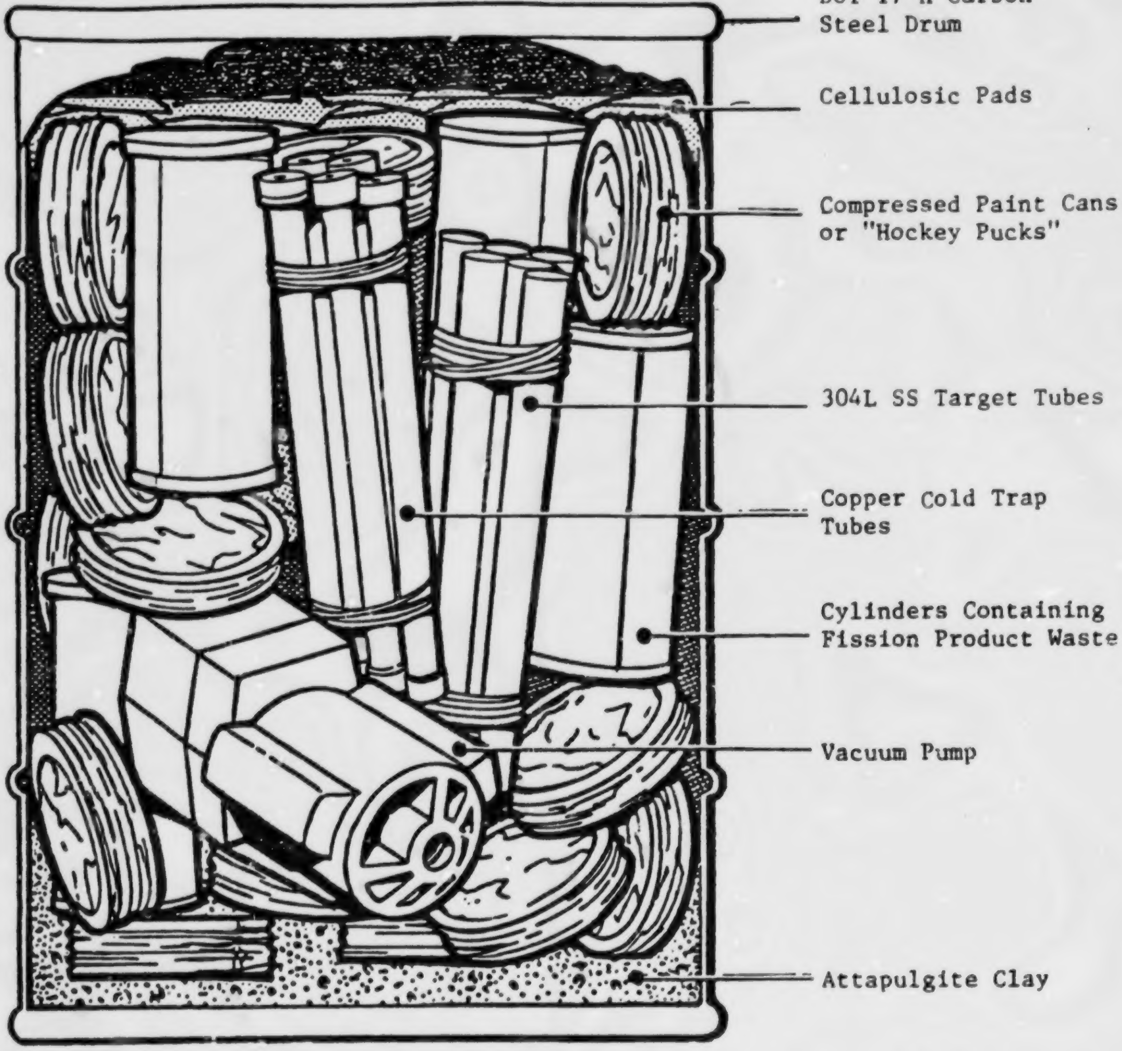
DOT 17 H Carbon
Steel Drum
Cellulosic Pads
Compressed Paint Cans
or "Hockey Pucks"
304L SS Target Tubes
Copper Cold Trap
Tubes
Cylinders Containing
Fission Product Waste
Vacuum Pump
Attapulgite Clay

Figure A.4. Typical Materials Comprising High Activity Radioisotope Production Waste Stream

plate. Liquid is introduced into the cylinder along with sufficient cement to (1) eliminate the potential for free liquid, and (2) ensure that the Pu-239 loading in the cylinder is less than 10 nCi/gm. After about an hour, a threaded 304 SS plug wrapped with Teflon tape is screwed into the nozzle. The filled container is then allowed to cure for about a week within the hot cell.

A summary of monthly waste shipments is given below for the years 1982 and 1983. A more detailed summary of shipments for 1983 is given in Table A-16.

	1982		1983	
Month	Activity (Ci)	Volume (ft^3)	Activity (Ci)	Volume (ft^3)
January	3,946.72	60.	215.90	90.
February	4,117.30	60.	183.30	75.
March	533.34	15.	186.07	75.
April	-	-	500.67	105.
May	1.26	885.5	721.75	90.
June	-	-	-	-
July	-	-	825.62	1,170.5
August	-	-	969.84	953.5
September	236.90	60.	772.26	90.
October	216.46	975.5	1,076.16	105.
November	191.31	60.	920.61	90.
December	141.43	75.	803.19	75.
Total:	9,384.72	2,191.	7,175.37	2,919.

As can be seen, a period of 5 months elapsed in 1982 in which no high activity wastes were shipped. This was the result of a change in company policy in which on-site interim storage was instituted for the high activity waste, resulting in a reduction in the shipped radioactivity inventory by about a factor of 20.

Based on these data, the following annual waste projections are made:

Waste Stream	Description	Volume (m^3/yr)	Ave. Concentration (Ci/m^3)
N-ISOPROD	High-activity process waste	30	2.45E+2
N-ISOTRSH	Low-activity general trash	52.2	6.50E-2

The data indicates that the average concentration for the low activity waste stream does not vary appreciably. More variation is seen for the high activity stream--i.e., 1983 data indicates a range from 1.10 to 16.05 Ci/ft^3 (39 to 567 Ci/m^3).

Table A-16. Summary of Isotope Production Facility Waste Shipments for 1983

Month	Activity (Ci)	Volume (ft^3)	Conc. (Ci/ft^3)
J	49.14	15	3.28
J	18.26	15	1.22
J	26.46	15	1.10
J	45.32	15	3.02
J	45.34	15	3.02
J	31.38	15	2.09
F	24.57	15	1.64
F	23.31	15	1.55
F	35.28	15	2.35
F	47.86	15	3.19
F	52.28	15	3.49
M	33.45	15	2.23
M	30.22	15	2.01
M	22.70	15	1.51
M	39.05	15	2.60
M	60.65	15	4.04
A	32.90	15	2.19
A	76.93	15	5.13
A	102.20	15	6.81
A	114.79	15	7.65
A	59.35	15	3.96
A	55.25	15	3.68
A	59.25	15	3.95
M	114.70	15	7.65
M	74.50	15	4.97
M	100.85	15	6.72
M	139.97	15	9.33
M	165.17	15	11.01
M	126.56	15	8.44
J	-	-	-
J	117.32	15	7.82
J	2.60	1050.5	2.47 E-3
J	77.02	15	5.13
J	152.27	15	10.15
J	64.45	15	4.30
J	73.88	15	4.93
J	140.00	15	9.33
J	121.09	15	8.07
J	76.99	15	5.13
A	73.84	15	4.92
A	67.48	15	4.50
A	114.79	15	7.65
A	146.30	15	9.75
A	133.58	15	8.91
A	121.09	15	8.07
A	1.26	833.5	1.51E-3
A	108.53	15	7.24
A	202.97	15	13.53
S	190.35	15	12.69
S	190.37	15	12.69
S	95.11	15	6.34
S	67.19	15	4.48
S	127.05	15	8.47
S	102.19	15	6.81
O	190.38	15	12.69
O	240.76	15	16.05
O	104.78	15	6.99
O	139.99	15	9.33
O	124.88	15	8.33
O	159.19	15	10.61
O	116.18	15	7.75
N	171.45	15	11.43
N	124.15	15	8.28
N	145.66	15	9.71
N	133.54	15	8.90
N	114.50	15	7.63
N	231.31	15	15.42
D	123.74	15	8.25
D	184.07	15	12.27
D	152.64	15	10.18
D	184.05	15	12.27
D	158.69	15	10.58

A.5.2.2 Waste From Large Tritium and Carbon-14 Manufacturers

This group of waste streams includes wastes generated from large-scale manufacturing operations principally producing pharmaceuticals and other labeled products. Product users include universities, hospitals, pesticide manufacturers, pharmaceutical firms, and others conducting research and test programs. The principal radioisotopes of concern appear to be tritium and carbon-14, although other shorter lived isotopes such as sulfur-35 also appear to be involved to some degree. A number of waste streams having different physical and chemical characteristics are generated. These are primarily generated in NRC Region I (NE region) and NRC Region III (MW region) as discussed below.

NE Region Waste. Wastes from the NE region can be separated into nine waste streams, of which six can be categorized as low activity waste streams and three can be categorized as high activity waste streams. The low activity waste streams are listed below.

Low Activity Streams

Symbol	Waste Stream	Volume Rate (m^3/yr)	Concentration (Ci/m^3)
N-NECOTRA	Compactible trash	158.	0.54
N-NEABLIQ	Absorbed organic liquid	9.9	74.6
N-NESOLIQ	Solidified aqueous liquid	46.6	26.3
N-NEVIALS	Rejected product vials	0.9	47.5
N-NENCGLS	Noncompactible glass	43.7	20.0
N-NEWOTAL	Noncompactible wood and metal	16.7	0.61

The volume rates and concentrations are given in an as-generated form. Packaging for shipment would further increase volumes (and reduce concentrations) by the following factors:

Waste Stream	Volume Increase Factor
N-NECOTRA	1.0*
N-NEABLIQ	4.1
N-NESOLIQ	1.4
N-NEVIALS	3.0
N-NENCGLS	1.0
N-NEWOTAL	1.0

*VRF for this waste stream = 2.0

The N-NECOTRA waste stream consists of compacted solid laboratory trash as well as compacted plastic and glass containers (e.g., 1-gallon plastic jugs) which at one time held radioactive material. A somewhat arbitrary volume reduction factor of 2 is assumed for this waste stream. This waste stream contains organic chemicals, as does the N-NEABLIQ waste stream. In this case, organic liquids are absorbed onto an attapulgite clay absorbent within a 30-gallon drum. This waste drum is placed within a 55-gallon drum with the remaining space within the 55-gallon drum filled with absorbent. This results in a total volume increase factor of about 4.1. The N-NESOLIQ waste stream consists of aqueous liquid wastes which are solidified within a mixture of cement and clay. A volume increase factor of about 1.4 is assumed.

The N-NEVIALS waste stream consists of glass product vials which cannot be sold for some reason, such as failure in fully meeting product specifications or exceeding the product shelf life. Typically, each vial contains about 1/20 mL of liquid, and is packaged using absorbents in a similar manner as liquid scintillation vials. A volume increase factor of 3 is therefore assumed.

The N-NENCGLS waste stream consists of glassware for which compaction is conservatively not performed due to concerns regarding the volatility of the tritium contamination. This waste stream is a composite of separate waste streams consisting of laboratory glass and glass slides used in thin layer chromatography.

The last waste steam, N-NEWOTAL, consists of contaminated noncompactible wood and metal (pipes, ductwork, etc.).

Three high activity waste streams are also generated. Two contain tritium, in one case in a gaseous form and in the other case in an absorbed organic liquid form. The third waste stream consists of absorbed organic liquids containing carbon-14. These waste streams are summarized below:

		As-Generated*		As-Shipped		
Symbol	Waste Stream	Vol. (m^3/yr)	Conc. (Ci/m^3)	Vol. (m^3/yr)	Conc. (Ci/m^3)	VIF**
N-NETRGAS	Tritium gas	1.08E-1	4.62E+4	5.78E-1	8.65E+3	5.35
N-NETRILI	Absorbed tritiated liquid	9.72E-1	4.62E+4	5.20E+0	8.65E+3	5.35
N-NECARLI	Absorbed C-14 liquid	4.86E-2	4.11E+2	2.60E-1	7.69E+1	5.35

*Over volume of primary container.
**Shipment volume increase factor.

Packaging is via high integrity containers developed especially for these three waste streams. Each high integrity container consists of separate concentric containers. The primary container consists of a 4-inch diameter aluminum (606IT-6) canister having an approximate length of 31-½ inches and an approximate wall thickness of 1/8 inch. Each primary container contains a number of sealed thick-walled (5/32 inch) pyrex glass bulbs about 120 mL in volume. Within each bulb a small amount of organic solvent and/or aqueous liquid is absorbed onto an absorbent principally consisting of attapulgite clay. In the case of gas, no such absorbent is used. Each glass bulb placed within the primary containers is separated from other bulbs using cardboard spacers. Void spaces within the

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primary container are filled with absorbent, which further assures against possible damage to the glass bulbs. The ends of the primary container are welded on (heliarc) and the welds visually inspected and tested using ultrasound.

The secondary container consists of a stainless steel (304) pipe having an inside diameter of 10-3/4 inches, a length of 32-½ inches, and a pipe thickness of 1/4 inch. Three of the primary containers are placed within each stainless steel container, with the void spaces within the stainless steel container filled with absorbent. The end pieces are welded on and again visually inspected and tested using ultrasound. Two stainless steel containers are then placed within a 55-gallon drum with the void space filled with absorbent and the drum lid welded on.

MW Region Waste. This group consists of four waste streams generated from syntheses of carbon-14 and tritium-labeled organic compounds. These waste streams are summarized below:

		As-Generated		As-Shipped		
Symbol	Waste Stream	Vol. (m^3/yr)	Conc. (Ci/m^3)	Vol. (m^3/yr)	Conc. (Ci/m^3)	VIF*
N-MWTRASH	Laboratory trash	6.3	1.84	6.3	1.84	1.0
N-MWABLIQ	Absorbed organic liquid	0.38	207.	2.1	37.7	5.5
N-MWSOLIQ	Solidified aqueous liquid	0.36	1,014.	0.5	724.	1.4
N-MWWASTE	Miscellaneous waste	1.5	35.0	1.5	35.0	1.0

Based on disposal records, the N-MWTRASH waste stream consists of miscellaneous materials (lab glassware, thin layer chromatography plates, etc.) typically placed into paint cans or used solvent cans which are then bagged in polyethylene and placed into 55-gallon drums. Smaller quantities of absorbents from chromatography columns are also placed into the drums along with used filters. The waste contains organic chemicals.

The N-MWABLIQ waste stream consists of organic liquids which are absorbed onto diatomaceous earth. A typical procedure is to fill a one-gallon plastic jug half full of absorbent, and then add waste liquid until the liquid level overtops the top level of the absorbent. The jugs are then completely filled with absorbent, capped, double bagged in plastic, and placed in a 55-gallon drum packed with additional absorbent. Up to 20 jugs can be placed in a single drum, which is equivalent to a packaging volume increase factor of 5.5. Cement is used to solidify the N-MWSOLIQ waste stream.

The N-MWWASTE waste stream principally consists of an organic sludge residue left on material such as paper, plastic or polyvinylchloride after evaporation of small amounts of organic liquids. About 90-95% of the waste material volume is plastic paper and most of the rest is glass. This material is packaged in

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drums along with absorbent material. Smaller quantities of absorbed liquids are also generated as are (occasionally) contaminated stainless steel or aluminum gas chromatography columns.

A.5.2.3 Sealed Source Manufacturing Waste

Waste from this source is difficult to project for a number of reasons. There is currently a limited amount of data on source manufacturers--i.e., on production rates, present and future isotope use, manufacturing process efficiency, and so forth. Waste characteristics (volumes, radionuclide quantities and concentrations, etc.) appear to vary considerably from one manufacturer to another. Source manufacturers are licensed by both NRC and Agreement States.

Given the above, the best approach would be to characterize the waste from each manufacturer individually. Insufficient resources, however, are available at this time to accomplish this. A limited characterization is thus performed which will concentrate on waste generated from three of the largest facilities. One facility is believed to be the principal manufacturer of americium-241 smoke detector foils. This is believed to account for most of the waste from sealed source manufacture that exceeds Part 61 Class C concentrations. (Most of the waste from sealed source manufacture is believed to fall within the Part 61 classification system.)

Facility 1. The majority of waste (exceeding Class C concentrations) from manufacture of americium sources is believed to orginate in NRC Region I. About 250 ft^3 (7.1 m^3) of waste has been accumulated as of the end of 1984, and about 100 ft^3 (2.8 m^3) is being generated per year (Refs. 47, 48).

This waste stream, termed N-SORMFG1, is therefore assumed to consist of 7.1 m^3 of manufacturing waste through 1984. This waste is assumed to be generated in NRC Region I at a rate of 2.8 m^3/yr.

Facility 2. The second facility examined is located in Region V. This facility generates waste from three principal sources: hot cell activities, liquid waste evaporator operations, and a research reactor (currently shut down). The hot cell activities appear to generate most of the waste and consist mainly of examination work on reactor components, the fabrication of Co-60 and Cf-252 sources, and the manufacture of Xe-133, Cl-36, and C-14 radiopharmaceuticals. Some of the waste from examination of reactor components has a very high transuranic component and is currently being stored on site. This transuranic-contaminated waste is considered in more detail in Section A.3.2.2. The on-site evaporator is used to treat liquid waste generated from on-site as well as off-site sources. Off-site sources include a nuclear power reactor having inadequate on-site water treatment capability. The research reactor is currently shut down, but may start up again in the future (Ref. 36).

A summary of waste shipments for the years 1980 through 1983 is given below:

	1980	1981	1982	1983	Average
Total Volume Shipped (ft^3)	10,400	12,900	6,500	5,900	8,925
Total Number of Shipments	12	32	15	19	19
Activity Total (Ci)	651	1,216.4	639.0	1,643.2	1,037.4[b]
Co-60	521	1,146	515.8	1,095.2	819.5
Cs-137	130.5	70.2	123.2	405.4	182.3
Other			3.5 mCi (U-235)	100 Ci (Sb-124) 42.46 Ci (C-14) 127 mCi (U-235)	
Co-60/Cs-137 ratio	3.99	16.32	4.19	2.70	4.50
Total Packages[a]					
Drums (55-gallon)	56	96	108	134	98.5 (724 ft^3)
Boxes	90	106	54	44	73.5 (8100 ft^3)
Casks (11.5 ft^3)	11	11	4	9	8.75 (101 ft^3)

[a] The vast majority of the waste activity is concentrated in the cask packages. The 55-gallon drums and boxes contain essentially trace amounts.

[b] Implies an "average" concentration of about 0.1 Ci/ft^3.

"Casks" are actually 11.5 ft^3 17H drums, each containing two inner perforated carbon steel baskets. The baskets are held within the drums away from the drum walls, so that after the waste baskets are placed within the drums, the space around the baskets as well as the interstitial voids within the baskets are filled with cement grout. The waste is generated as a result of hot cell operations and consists of liquid waste (about 30 gal/yr) solidified with cement in one-gallon cans, compacted one-gallon cans filled with dry solid waste, glass, and irradiated metals. The density of the waste is about 0.5 g/cm^3 prior to grouting and about 2.35 g/cm^3 after grouting. Typical total weights of a cement grouted drum range from 1,600 to 2,300 lbs, of which about 300 lbs is waste. The composition of the waste is about 40% metal (ferrous, some aluminum, <10% lead), 10% glass, and 50% paper and plastic.

The fifty-five gallon drums and boxes are generated as part of "support" activities and comprise about 70% of the waste by volume; essentially all would be classified as Class A waste under 10 CFR Part 61. Wastes placed in drums include dewatered resins, solidified liquids, and non-compactible trash. A representative radioactivity loading for dewatered resins is about 0.3 μCi of Co-60 per cubic centimeter. Liquids include wastes from cleanup activities, fuel rod decrudding, hot water leach-outs from fuel elements, and evaporator bottoms. Generation of liquid wastes outside hot cells is approximately 100,000 gal/yr. Much of this is sent to the waste evaporator from which about 100-125 55-gallon drums are generated per year containing cement solidified bottoms. Non-compactible items include equipment components, glass, etc. A typical waste package weighs 600-700 pounds. Waste boxes contain compactible lab wastes and average about 2000 lbs apiece. Each box is constructed of plywood and generally contains about 112 ft^3 of waste. Other minor sources of waste consist of vacuum pump oil contaminated with Xe-133 (about 25 gal/yr) and liquid scintillation vials (about one 55-gallon drum every 2 years).

The above information implies a total waste generation rate of about 253 m^3 per year, of which about 8.1% consists of 55-gallon drums, 90.8% consists of wooden boxes, and about 1.1% consists of the grouted "casks" (11.5 ft^3 drums). Reference 36 further implies an average gross concentration of about 1.6E-3 Ci/ft^3 in the waste drums and boxes (i.e., 3.885 Ci in 2,438.5 ft^3). Applying this concentration over the average waste volume for 1980-1983 implies an average gross concentration of the "cask" waste of about 10.1 Ci/ft^3. A review of 1983 shipment records for four grouted casks indicates a range of 4.5 - 16.2 Ci/ft^3 (average 8.6 Ci/ft^3).

Thus, a single waste stream (N-SORMFG2) is assumed which represents the heterogeneous mixture of waste types discussed above. This waste stream is assumed to be generated in NRC Region V at an average rate of 253 m^3/yr. About 99% of the waste is assumed to be of low activity--averaging 5.65E-2 Ci/m^3--while 1% of the waste (the "cask" waste) is assumed to be of high activity--averaging about 3.51E+2 Ci/m^3 (10 Ci/ft^3). This activity is assumed to consist primarily of Co-60 and Cs-137 at an average Co-60/Cs-137 ratio of about 4.5.

Facility 3. This facility is located in NRC Region III and produces four general categories of radioactive products, including radiopharmaceuticals, surgical implants, labeled compounds for research, and radioactive sources for static eliminators and other applications. A summary of radioactive products or services is given below (Ref. 49).

Isotope	Typical Waste Package Classification	Product/Service
Sr-90	B, C	Sources (labeled ceramic microspheres)
Cs-137	B, C	Sources (labeled ceramic microspheres)
Ra-226	A	Disposal service for new cesium users*
Yb-169	A	Yb-169 DTPA (diethylenetriaminepentaacetic acid)
Tc-99m	A	Human serum albumin microspheres labeled with Tc-99m
I-125	A	Stock microspheres
Ce-141	A	Stock microspheres
Cr-51	A	Stock microspheres
Sr-85	A	Stock microspheres
Nb-95	A	Stock microspheres
Sc-46	A	Stock microspheres
Po-210	A	Static eliminators
Yb-169	A	Customer requested microspheres
Fe-59	A	Customer requested microspheres
Ca-45	A	Customer requested microspheres
In-114m	A	Customer requested microspheres
I-125	A	Seeds for surgical implants (iodine plated on silver wire)

*Service no longer provided.

As can be seen, the products and isotopes are quite varied, and so the characteristics of the waste produced during facility operations are also quite varied. However, two general waste types can be considered. One is a high activity waste stream which may be classed as either Class B or Class C under 10 CFR 61, and which is contaminated mostly with Cs-137 (and to a lesser extent with Sr-90). This waste stream is designated N-SORMFG3 and mainly results from production of Cs-137 miscrospheres. The second waste stream, designated N-SORMFG4, is of low activity and may be contaminated with any of the radioisotopes discussed above. This waste stream is a general aggregation of waste generated during balance-of-plant operations.

Cesium as a CsCl powder is obtained from ORNL and is incorporated into ceramic microspheres averaging 60 to 100 μm in diameter. The microspheres are placed in stainless steel (or k-monel) cylinders of various lengths and diameters, which are then closed by welding, cleaned with trisodium phosphate, and leak tested. Leaking sources are rejected and become a waste product, along with contaminated equipment, paper, plastic vials, HEPA filters, evaporates of aqueous solutions, glassware, metal, failed microspheres, contaminated glycerol, and other scrap materials resulting from hot-cell fabrication activities.

Most of this waste (about 50% plastic, 10% HEPA filters, and 40% glass, paper, metal and waste microspheres) is placed in Nalgene containers (low density polyethylene) which are then packaged in 30-gallon steel drums which are lined with 2 to 3 inches of lead (inside volume ranging from 1.6 to 2.1 ft^3). Present practice is to place no more than 500 Ci of Cs-137 in a container. From 1 to 10 such waste drums (average about 6) are produced per year. This results in about 0.35 m^3 of this waste per year, prior to placement in the lined drums

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which increase the disposal volume by a factor of about 2. In the past, wooden overpacks have also been used which further increased the disposed volume by about 3.7.

Occasionally a large piece of equipment is also placed within a lead-lined drum, without, however, first placing the equipment in a Nalgene container. Reject Cs-137 sources, as well as sources occasionally returned to the company, are placed within lead pigs and then into a lead-lined drum. One such drum filled with Cs-137 sources in lead pigs is produced every 2-3 years. Thirty-gallon drums lined with 3 inches of concrete are also occasionally used for some wastes such as those produced from cleaning the sources after fabrication.

Wastes from other activities are disposed in wooden crates, 55-gallon carbon steel drums, or 30-gallon concrete-lined (3-inch lining) drums. Most of the waste is generated as part of production of Po-210 microspheres for static eliminators. Other wastes are generated as part of use of H-3 and C-14 labeled radiotracers, production of surgical implant seeds containing I-125, production of gamma labels in carbonized resin beads (a multitude of isotopes), production of radiopharmaceutical kits containing Yb-169 DTPA solution, and production of Tc-99m labeled human serum albumin microspheres. Sealed sources are also constructed containing Sr-90 microspheres. Wastes from these operations are similar in physical characteristics to those from Cs-137 operations, but of generally lower activity and containing very short-lived radiocontaminants. Such wastes include reject microspheres and sources, paper, glass, plastic, metal and other trash. This waste is estimated to be generated at a rate of about 160 m^3/yr (Ref. 49).

A summary of the as-generated volumes and gross concentrations projected for the N-SORMFG3 and N-SORMFG4 waste streams is provided below:

Waste Stream	Volume (m^3/yr)	Concentration (Ci/m^3)
N-SORMFG3	0.35	6.0E+3
N-SORMFG4	160	1.11E+0

A.5.2.4 Waste From Small Tritium Manufacturers and Users

Tritium is a radioisotope very commonly used by NRC and Agreement State licensees, often in large quantities. In addition to applications in biological research and medicine, it is used in a wide variety of products such as illuminators. There are also a large number of NRC and Agreement State licensees that may generate such waste, which makes projecting volumes and activities of wastes containing large amounts of tritium difficult. This is contrasted by the previous section where a very small number of waste generators are involved.

Waste streams considered in this section include high activity solidified and absorbed liquids, gases, high activity liquid scintillation vials, tritium contained in paint or as a plating, and tritium contained in metal or foils. The most common application for the latter waste type (metal or foils) is as accelerator targets.

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Accelerator targets are used to produce radionuclides by direct bombardment with charged particle beams or by indirect reactions of the target fragments with other materials. Accelerator targets are also used to study nuclear reactions and to produce and study the properties of various subatomic particles. Targets from institutional sources are also included in this waste stream. Spent targets are commonly made of titanium foils containing absorbed tritium.

Projections are made based on a review of disposal records for the three operating low-level waste disposal sites for the year 1982. In the review, waste generators were first identified that generated and shipped at least 100 mCi of tritium in 1982. Then, the total activity of tritium that was shipped was compared with a "theoretical" total activity based upon the shipment volume and using the projected low activity tritium concentrations for industrial and institutional waste generators. That is, if a waste generator was identified as an institutional waste generator, an "average theoretical" tritium concentration was estimated based on concentrations in the I-COTRASH, I-ABSLIQD, I-LIQSCVL, and I-BIOWAST waste streams. This average theoretical tritium concentration was multiplied by the 1982 waste volume for a particular waste generator and the resulting theoretical tritium activity compared to the actual tritium activity. If the actual tritium activity exceeded the theoretical tritium activity by a significant margin (i.e., by about a factor of 10), the disposal records for the waste generator were examined further to identify specific waste packages containing large (at least 100 mCi) quantities of tritium. A similar procedure was used for industrial waste generators except that the N-LOTRASH and N-LOWASTE waste streams were used. (The large industrial generators considered in the above Section A.5.2.2 were not included in this survey.)

Using the above technique, waste packages containing at least 100 mCi of tritium were identified. A total of 750 waste packages were so identified--generally 30-gallon or 55-gallon drums. Each waste package was then characterized in terms of volume, tritium concentration, waste form, and state in which the waste was generated. This sometimes proved difficult, in that the waste descriptions in some waste shipment papers for some waste generators were insufficient to clearly identify a particular waste form. In other cases, a waste broker rather than the generator was identified on the manifest. In this case, the State in which the waste broker is located was used.

This survey identified 4511.6 ft^3 of waste in 750 waste packages, with each waste package containing at least 100 mCi of tritium, and having a total tritium activity of 40,016 Ci. Of this waste, the waste form for 279.02 ft^3 (59 waste packages, 774 Ci of tritium) could not be clearly identified. This left 4252.58 ft^3 of waste and 691 waste packages.

The results of the survey are summarized in Table A-17. Eight waste forms are identified: paint and plating, gas, liquid scintillation vials, absorbed scintillation liquid, absorbed aqueous liquid, solidified liquid, trash, and foils (including gas absorbed in metals). For each of the eight waste forms the table lists the total waste volume, the concentration distribution, the regional distribution, and the total number of waste packages.

The activity distributions were generated by placing the volume and concentration of each waste package into one of six concentration ranges. The total volume of waste falling into each concentration range was determined, as was

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the average of all the waste concentrations in the concentration range. The table thus lists, for each waste form and concentration range, the percent of the waste volume falling into the range plus the average concentration (in parentheses) across the range. (For example, 11.5% of the disposed volume of absorbed aqueous liquids falls into a concentration range of between 1 and 10 Ci/ft^3, and the average concentration of all absorbed aqueous liquids in this concentration range is 2.42 Ci/ft^3.) The regional distribution gives the percent of the waste volume that is generated in each NRC region for each waste form.

Based on the data in Table A-17, projections of waste from small tritium manufacturers and users were made. The total number of waste streams considered was reduced to six by combining scintillation liquids and vials into one stream, and solidified and absorbed liquids into another stream. (Absorbed and solidified liquids, for example, can be considered to be the same type as generated waste--a liquid waste stream--but are packaged for shipment in different ways.)

The resulting six waste streams and assumed annual generation volumes are as follows:

Waste Stream	Description	Volume (m^3/yr)
N-TRIPLAT	Tritium contained in paint or as a plating	4.4
N-TRITGAS	Tritium disposed in a gaseous form	24.6
N-TRISCNT	Tritium contained in scintillation liquid	1.3
N-TRILIQD	Tritium contained in aqueous liquid	7.1
N-TRITRSH	Tritium contained in or contacting miscellaneous trash (e.g., plastic, paper, glass)	23.9
N-TRIFOIL	Tritium contained in or absorbed into metal (e.g., foils)	10.6

In projecting waste volumes, the total volume waste identified in 1982 as containing at least 100 mCi of tritium was distributed among the six waste streams according to the volume percentages listed in Table A-17, and converted to m^3. The volumes of the two liquid waste streams (N-TRISCNT and N-TRILIQD) are furthermore given in an as-generated (untreated) form. This is done since there are a number of processing and packaging options that can be considered for such wastes. (Note that processing and packaging options will alter the waste volumes for these streams. Shipment as an absorbed liquid, for example, will increase volumes by about a factor of 3). The other four waste streams are more difficult to characterize and are thus given in an as-packaged waste form.

The waste streams are furthermore characterized in terms of their concentration and regional distribution in Table A-18. As before, the table lists for each waste stream the volume percent falling into each concentration range, as well as the average concentration across the concentration range (in parentheses). The weighted average concentration is the sum (over all ranges) of the average concentration in each range multiplied by the volume fraction for that range. The regional distribution gives the volume percent assumed to be generated in each region for each waste form.

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Table A-17. Summary of Survey of High Activity Tritium Waste Streams from Small Generators

	Paint and Plating	Gas	Abs. LSC Vials	Abs. Scint. Liquid	Solid. Aqueous Liquid	Absorbed Aqueous Liquid	Trash	Foils, Gas Abs. in Metal	Totals
Total Volume (ft^3)	145.15 (3.4%)	817.88 (19.2%)	29.19 (0.7%)	105 (2.5%)	1850.89 (43.5%)	137.74 (3.2%)	795.66 (18.7%)	351.07 (8.3%)	4252.58
Percent and Ave. Conc. (Ci/ft^3) in Conc. Range									
$.01 \leq .1$ Ci/ft^3			51.4 (2.67E-2)	28.6 (3.70E-2)	34.9 (2.59E-2)	5.4 (1.34E-2)	22.7 (3.36E-2)	3.3 (5.76E-2)	
$.1 < 1$ Ci/ft^3	72.3 (4.49E-1)	2.3 (6.69E-1)	46.3 (4.10E-1)	64.3 (5.49E-1)	43.7 (1.98E-1)	27.2 (5.49E-1)	54.9 (2.88E-1)	8.9 (2.79E-1)	
$1 \leq 10$ Ci/ft^3	13.8 (2.59)	44.1 (5.30)	2.3 (7.46)	7.1 (1.16)	11.5 (2.42)	27.2 (4.85)	17.5 (2.2)	8.8 (2.42)	
$10 \leq 100$ Ci/ft^3	10.3 (2.98E+1)	48.7 (1.65E+1)			9.5 (2.37E+1)	38.1 (3.07E+1)	4.9 (2.90E+1)	78.4 (1.33E+1)	
$100 \leq 1,000$ Ci/ft^3	3.6 (2.81+E-2)	4.8 (1.34+2)							
$1,000 \leq 10,000$ Ci/ft^3						2.0 (1.98E+3)		0.6 (1.90+3)	
NRC Region (%)									
I	6.3	56.9	22.9	14.3	8.6	81.7	55.6	89.7	36.0 (1525.46 ft^3)
II	5.2	0	0	85.7	0	0	28.3	9.7	8.4 (356.4 ft^3)
III	26.6	2.4	51.4	0	2.4	10.9	13.2	0	5.6 (238.07 ft^3)
IV	62.0	39.7	25.7	0	0	5.4	1.9	0	10.5 (444.8 ft^3)
V	0	1.0	0	0	89.0	2.0	1.0	0.6	39.4 (1667.85 ft^3)
No. of Data Points*	16	166	8	14	260	24	151	52	691

*The total (small generator) data base for 1982 included 4511.60 ft^3 of waste in 750 waste packages, with each waste package containing at least 100 mCi of tritium. Of this waste, the waste form of 279.02 ft^3 (59 waste packages) could not be clearly identified. This left 4252.58 ft^3 of waste (691 waste packages).

Table A-18. Summary of Waste Streams From Small Tritium Manufacturers and Users

	N-TRIPLAT	N-TRITGAS	N-TRISCNT	N-TRILIQD	N-TRITRSH	N-TRIFOIL
Stream Volume (m^3)*:	4.4	24.6	1.3	7.1	23.9	10.6
Concentration Range (Ci/m^3)**:						
$0.35 \leq 3.5$			16.8 (2.33)	22.2 (1.96)	22.7 (1.19)	3.3 (2.03)
$3.5 \leq 35$	72.3 (1.59E+1)	2.3 (2.36E+1)	33.5 (1.02E+1)	40.2 (1.26E+1)	54.9 (1.02E+1)	8.9 (9.85)
$35 < 350$	13.8 (9.15E+1)	44.1 (1.87E+2)	49.2 (6.99E+1)	20.2 (1.06E+2)	17.5 (7.77E+1)	8.8 (8.55E+1)
$350 \leq 3,500$	10.3 (1.05E+3)	48.7 (5.83E+2)	0.5 (7.91E+2)	13.2 (1.19E+3)	4.9 (1.02E+3)	78.4 (4.70E+2)
$3,500 \leq 35,000$	3.6 (9.92E+3)	4.8 (4.73E+3)		3.9 (5.97E+3)		
35,000+				0.3 (9.82E+4)		0.6 (6.71E+4)
Weighted Average (Ci/m^3)***:	4.89E+2	5.94E+2	4.22E+1	7.11E+2	6.94E+1	7.80E+2
NRC Region (%):						
I	6.3	56.9	16.2	18.6	55.6	89.7
II	5.2	0	67.1	0	28.3	9.7
III	26.6	2.4	11.2	3.6	13.2	0
IV	62.0	39.2	5.6	0.7	1.9	0
V	0	1.0	0	77.0	1.0	0.6

Also note that the N-TRIFOIL waste stream replaces the N-TARGETS waste stream assumed in reference 1.

A.5.2.5 Sealed Sources and Devices

Sealed sources are small, concentrated units of activity which are typically encapsulated to prevent leakage of the radioactive material. Low-activity sealed sources and foils are used as calibration and reference standards for many types of radiation detectors. High-activity sealed sources are used in neutron generators as well as medical and industrial irradiators. Other examples of industrial sources include density gauges, well logging sources, radiography sources, x-ray fluorescence tubes, static eliminators, and so forth. This waste stream includes industrial sealed sources as well as sealed sources from institutions.

A discussion regarding characteristics of sealed sources is provided in reference 50. From this reference it can be seen that there are two major groups of sealed sources and devices. These include: (1) discarded sealed sources, and (2) discarded sealed neutron sources. Volumes and quantities of such waste are difficult to project for a number of reasons.

First, sources are very small concentrated sources of activity--the average size of sealed sources above 10 curies is about 3 inches in length and 0.5 inches in diameter--and so it is not particularly meaningful to quantify sources in terms of volume.

Second, it is difficult to estimate the number and activity distribution of sealed sources which are currently in the possession of licensees, and which may eventually become candidates for disposal as waste. Some information is provided by Table A-19, which reformats data from reference 50 and gives the distribution of the maximum allowable activities for the 600+ source designs listed in the NRC sealed source registry. Included are sources exceeding 1 mCi of an isotope listed in Tables 1 and 2 of 10 CFR Part 61. This Table A-19 data needs to be used with caution, however, since it provides no information on the total number of sealed sources which have been sold. In addition, the manufacturing license for a particular source design specifies an activity limit rather than a required activity, and most sources that will be sold will have lower activities than the design limit (Ref. 50).

In general, it is believed that the activities of the majority of the sources sold trend toward the lower end of the possible spectrum. For example, Table A-19 lists 194 individual designs for Cs-137 sources ranging in activities from a few millicuries to several thousand curies. However, most Cs-137 sources are used in gauges and have sizes in the millicurie range. Relatively few large Cs-137 irradiation sources are sold, and of these large sources, most appear to be in a 100 to 1000 curie range, with the 500 curie size selling the most. The most popular Co-60 irradiation sources seem to be in a range of 30-100 curies, although they can range from 2,000 Ci to 6,000 Ci (Ref. 50).

One former manufacturer of large (encapsulated) sealed sources has estimated the total number of sources containing transuranic radioisotopes that has been sold during facility operation. This estimate is provided below and covers the years 1961 through 1984, during which time the manufacturer estimates to have

Table A-19. Distribution of Maximum Allowable Activities Among Sealed Source Designs

	Activity (Ci)								
Isotope	.001-.009	.01-.09	.1-.9	1.-9.	10.-90.	100.-900.	1000.-9000.	10,000.-90,000.	Totals
H-3	1* (3.7)	0	10 (37.0)	11 (40.7)	5 (18.5)	0	0	0	27
C-14	1 (50.0)	1 (50.0)	0	0	0	0	0	0	2
Co-60	2 (1.2)	24 (14.5)	27 (16.4)	23 (13.9)	25 (15.2)	31 (18.8)	14 (8.5)	19 (11.5)	165
Ni-63	2 (25.0)	6 (75.0)	0	0	0	0	0	0	8
Sr-90	5 (6.6)	13 (17.1)	28 (36.8)	27 (35.5)	3 (3.9)	0	0	0	76
Cs-137	5 (2.6)	42 (21.6)	62 (32.0)	41 (21.1)	17 (8.8)	16 (8.2)	11 (5.7)	0	194
Pu-238	0	1 (3.7)	9 (33.3)	7 (25.9)	10 (37.0)	0	0	0	27
Pu-239	0	0	0	0	2 (100.)	0	0	0	2
Am-241	5 (3.5)	25 (17.7)	47 (33.3)	49 (34.8)	15 (10.6)	0	0	0	141
Pu-238/ Be**	0	0	0	1 (33.3)	2 (66.7)	0	0	0	3
Am-241/ Be**	0	4 (25.0)	0	4 (25.0)	8 (50.)	0	0	0	16

*The first entry indicates the number of source designs observed in each range. The second entry, in parentheses, indicates the percentage.

**These are neutron sources; respectively: plutonium-beryllium and americium-beryllium sources.

accounted for about 70% of the market for encapsulated radiation sources containing these radioisotopes (Ref. 51).

Description	Approx. No. of Sources	Est. Total Content (Ci)	"Average" Content (Ci)
Am-241 gamma sources	800	400	0.5
Am-241-Be neutron sources	5,887	48,000	8.2
Am-241-B neutron sources	30	210	7.0
Am-241-F neutron sources	4	60	15.0
Am-241-Li neutron sources	121	1,200	9.9
Pu-239-Be neutron sources	466	1,400	3.0
Pu-238 heat sources	257	2,050	8.0
Pu-238 gamma sources	220	130	0.6
Pu-238-Be neutron sources	509	12,750	25.0
Pu-238-B neutron sources	3	50	16.7
Pu-238-F neutron sources	4	80	20.0
Pu-238-Li neutron sources	63	750	11.9
Np-234 threshold detector	916	-	-
Cf-252 neutron source	317	32	0.1
Total Am-241	6,842	49,870	7.3
Total Pu-238	1,056	15,810	15.0
Total Pu-239	466	1,400	3.0
Total Np-234	516	-	-
Total Cf-252	317	32	0.1

The number of sources that have been sold exceeds the number that are currently in the possession of individual licensees. This is because in many instances sources held by licensees may be returned to the manufacturer when they are no longer usable, which could arise from a number of occurrences. For example, the source might develop a leak, the licensee's manufacturing process might change, or the radioactive source activity might be depleted due to radioactive decay. In such cases it is frequently economically worthwhile to recover the radioactive material and recycle it into a new source.

A final consideration is that a sealed source does not become a source of waste until a decision is made to treat it as such, and this is rather unpredictable. If a source is no longer usable or recyclable due to economic or other nontechnical reasons, then it may possibly be placed in storage to await some change in the market or other climate. The question of when stored material becomes waste can only be answered on a case-specific basis.

Given the above, the following approach is taken to estimate sealed source waste streams. First, eleven sealed source waste streams are assumed which contain isotopes listed in Tables 1 and 2 of 10 CFR Part 61:

Designation	Description
N-TRITSOR	Tritium (H-3) source
N-CARBSOR	Carbon-14 source
N-COBLSOR	Cobalt-60 source
N-NICKSOR	Nickel-63 source
N-STROSOR	Strontium-90 source
N-CESISOR	Cesium-137 source
N-PLU8SOR	Plutonium-238 source
N-PLU9SOR	Plutonium-239 source
N-AMERSOR	Americium-241 source
N-PUBESOR	Plutonium-238 neutron source
N-AMBESOR	Americium-241 neutron source

Next, the following estimates are made for the number of sealed sources annually considered to be waste material:

Source	Number	Source	Number
N-TRITSOR	8	N-PLU8SOR	6
N-CARBSOR	4	N-PLU9SOR	4
N-COBLSOR	20	N-AMERSOR	8
N-NICKSOR	4	N-PUBESOR	6
N-STROSOR	8	N-AMBESOR	60
N-CESISOR	20		
			148

These estimates are subject to considerable uncertainties, and are made based on a number of considerations. For example, sources for which large numbers of designs are listed in Table A-19 would probably be sold in relatively large numbers. Conversely, a small number of designs implies a small number sold. In addition, a sense of the relative number of waste sources in existence can be acquired through examination of shipping records and through interaction with licensees.

In the case of Pu-239 sources, there is believed to be few still being held by licensees. These sources are currently not commonly made, and most of the Pu-239 sources that were made in the past are believed to have been collected by the federal government. For other transuranic sources, it is assumed that there are approximately 10,000 plutonium and americium sources being held by licensees, and that roughly one-quarter of these will be candidates for disposal within the next 30 years. This corresponds to an annual generation of roughly 80 waste sources containing Pu-238 or americium. The distribution among the four source types containing these isotopes is based on reference 51 and is assumed to roughly correspond to the relative number of sources sold.

Next, the regional distribution is assumed to be as follows: NRC Region I: 25%, NRC Region II: 25%, NRC Region III: 25%, NRC Region IV: 12.5%, and NRC Region V: 12.5%.

Finally, estimates are made of the activity distributions within the sealed source waste streams. Despite the uncertainties, a distribution approach is adopted, rather than use of simple averages, for some specific reasons. One reason is that sealed sources are known to exist in a wide range of activities. A distribution is therefore a better format for expressing the source activities than an average. As additional information is obtained, it can more easily be programmed into the analysis methodology. Another reason is that it better exercises the analysis methodology in that sources can be considered in more than one waste class.

Assumed activity distributions are listed in Table A-20. In general, the design distributions listed in Table A-19 are assumed to be applicable. This is believed to be very conservative since most sources contain considerably less activity than the particular design limit. These distributions are then scaled to an assumed weighted average activity for each waste stream. The average activity within each activity range is then estimated by solving the following equation for (a):

$$WA = f_1a + 10^1f_2a + 10^2f_3a + \ldots 10^{n-1}f_na \qquad \text{(A-1)}$$

where,

WA = weighted average activity (Ci) across the distribution;
f_n = fraction of activity in each activity range; and
a = average activity (Ci) within first activity range.

Estimates of the weighted-average activities are made using input from a number of sources, including references 50, 51, and 58. These estimates are listed below:

Source	Weighted Average (Ci)
N-TRITSOR	1
N-CARBSOR	0.01
N-COBLSOR	500
N-NICKSOR	0.01
N-STROSOR	1
N-CESISOR	100
N-PLU8SOR	4.6
N-PLU9SOR	3.0
N-AMERSOR	0.5
N-PUBESOR	23.5
N-AMBESOR	8.2

Table A-20. Assumed Activity Distributions for Sealed Source Waste Streams

Range (Ci)	N-TRITSOR	N-CARBSOR	N-COBLSOR	N-NICKSOR	N-STROSOR	N-CESISOR
0.001 $\le$ 0.01	3.7* (0.00159)*	50 (0.00182)	1.2 (0.00227)	25 (0.00129)	6.6 (0.00128)	2.6 (0.00157)
0.01 $\le$ 0.1		50 (0.0182)	14.5 (0.0227)	75 (0.0129)	17.1 (0.0128)	21.6 (0.0157)
0.1 $\le$ 1	37 (0.159)		16.4 (0.227)		36.8 (0.128)	32 (0.157)
1 $\le$ 10	59.2 (1.59)		13.9 (2.27)		35.5 (1.28)	21.1 (1.57)
10 $\le$ 100			15.2 (22.7)		3.9 (12.8)	8.8 (15.7)
100 $\le$ 1000			18.8 (227)			8.2 (157)
1000 $\le$ 10,000			20.0 (2270)			5.7 (1570)
10,000 $\le$ 100,000						
Wt. Ave (Ci)	1.0	0.01	500	0.01	1.0	100

*The first value gives the percentage of waste sources assumed to be in each activity range. The second value (in parentheses) gives the assumed average activity across the activity range.

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Table A-20 (continued)

Range (Ci)	N-PLU8SOR	N-PLU9SOR	N-AMERSOR	N-PUBESOR	N-AMBESOR
$0.001 \leq 0.01$			3.5 (0.00102)		
$0.01 \leq 0.1$	3.7 (0.0115)		17.7 (0.0102)		25 (0.0156)
$0.1 \leq 1$	33.3 (0.115)		33.3 (0.102)		
$1 \leq 10$	25.9 (1.15)	100 (3.0)	45.4 (1.02)	33.3 (3.36)	25 (1.56)
$10 \leq 100$	37.0 (11.5)			66.7 (33.6)	50 (15.6)
$100 \leq 1000$					
$1000 \leq 10,000$					
$10,000 \leq 100,000$					
Wt. Ave. (Ci)	4.6	3.0	0.5	23.5	8.2

A.5.2.6 Activated Metals (N-HIGHACT)

The high specific activity industrial waste stream is a generic stream that includes miscellaneous wastes of relatively high activity, which is arbitrarily defined as an activity that exceeds 3.5 Ci/m^3 or 0.1 Ci/ft^3. High specific activity industrial wastes are expected to include activated metal and equipment produced by accelerators, activated metal and equipment from research reactors and subcritical assemblies, and activated metal from neutron generators. Volumes and gross concentrations are obtained from reference 1, and are as follows:

Volumes (m^3)		Gross Average Concentration (Ci/m^3)
In 1980	Added Per Year*	
74.4	5	210

*Following the year 2000, annual waste generation is assumed to be constant.

Volumes and concentrations are given for waste as packaged for shipment. The regional distribution is assumed to be given as follows: NRC Region I: 31%, NRC Region II: 22%, NRC Region III: 27%, NRC Region IV: 8%, NRC Region V: 12%.

A.6 OTHER NON FUEL CYCLE WASTE SOURCES

Two other generators of non-fuel cycle of waste include licensees holding discrete sources of waste contaminated with radium-226, and U.S. military facilities.

Discrete sources of radium waste are considered in this report based upon a request to do so by the Conference of Radiation Control Program Directors (CRCPD) (Ref. 52). The terminology "discrete" is used here to identify waste in which the radium is concentrated in a relatively small volume of material, as opposed to waste in which the radium is concentrated in a large volume of material such as radium-contaminated dirt.

Based upon information provided by the CRCPD, there appear to be two principal discrete sources of radium waste. One consists of sealed sources such as old medical sources. For several years, such waste was collected and stored at an EPA laboratory in Montgomery, Alabama. Such waste is no longer being accepted for storage, however. The other source consists of ion-exchange media which may be generated in the future as a result of efforts to remove radium from groundwater supplies. A groundwater radium removal system is currently being tested by DOW chemical (Ref. 53).

Low-level radioactive waste generated at U.S. military facilities is occasionally shipped to commercial facilities for disposal. These facilities are not licensed by either NRC or Agreement States, although the waste must be treated, packaged and shipped in compliance with applicable regulations and license conditions in order to be accepted for disposal. The most significant fraction of this waste appears to be generated as part of maintenance of nuclear powered naval vessels. This waste consists of ion-exchange resins and other waste similar to that generated as part of routine operation of nuclear power plants. Other waste appears to consist of waste similar to that generated by other non-fuel cycle generators--for example, waste consisting mainly of source material, laboratory waste similar to that generated by institutions, sealed sources, and so forth.

A.6.1 Radium Contaminated Waste

Radium is a naturally occurring radioactive element which has been used in medical and industrial applications since the turn of the century. While there are several known isotopes of radium, the one that has the greatest utilization is radium-226, an isotope forming part of the uranium-238 decay scheme. The decay scheme for radium is shown below. (Throughout the remainder of this section, the term "radium" is used as a general reference to radium-226 and its daughters.)

Radionuclide	Common Name	Half-Life	Radiation Emitted
Ra-226	Radium	1602 y	α, γ
Rn-222	Radon	3.83 d	α, γ
Po-218	Radium A	3.05 m	α, β
Pb-214	Radium B	26.8 m	β, γ
Bi-214	Radium C	19.7 m	α, β, γ
Po-214	Radium C'	1.64E-4 s	α, γ
Pb-210	Radium D	20.4 y	β, γ
Bi-210	Radium E	5.013 d	β
Po-210	Radium F	138.4 d	α, γ
Pb-206	Radium G	stable	none

Radium acts in a chemically similar manner to calcium, strontium, and barium, and thus tends to concentrate in the bone. Since it is an alkaline metal that reacts with nitrogen, in commercial use it is principally in the form of a salt. In most commercial uses, radium is used in equilibrium with several short-lived daughters, including radon-222 through polonium-214 (Radium C'). Starting with pure radium-226, equilibrium is reached in about 30 days. The next decay daughter, lead-210 (Radium D), requires about 100 years to reach equilibrium with radium-226. Radium D and Radium E have been most commonly used as sources of beta radiation (Ref. 54).

The remainder of this section considers radium waste in two general groups. The first group consists of several waste streams in which radium exists as a small discrete source--i.e., a discarded sealed source device formally used for medical or industrial applications. The second group consists of ion-exchange resins used to remove radium from groundwater.

A.6.1.1 Radium Sources

Available Data

For a number of years the U.S. Department of Health, Education, and Welfare, in conjunction with the U.S. Environmental Protection Agency (EPA), conducted a program for collection and storage of radium waste. The radium waste consisted of about 150 grams of radium contained within better than 13,000 sources and was stored at EPA's Eastern Environmental Radiation Facility in Montgomery, Alabama. A summary of the receipts at the facility for the years 1974 through 1981 is given below (Ref. 55).

Year	No. of Sources	Activity (Ci)
1974	620	3.5
1975	850	4.8
1976	650	4.8
1977	1,100	11.0
1978	1,050	9.5
1979	700	5.8
1980	2,500	19.0
1981	1,300	7.0
Total:	8,770	65.4 Ci
Ave:	1,096	8.2

This program was eventually discontinued, and all the radium waste that had been collected was removed from the facility and disposed in 1983 in a low-level waste disposal site. The records obtained from this storage operation provide a means of characterizing the radium waste into six separate radium waste streams. These six waste streams include four medical and two non-medical waste streams.

Medical sources are relatively easy to characterize, and can be grouped into four basic waste streams: needles, cells, plaques, and nasopharyngeal applicators. Typical medical source designs are illustrated in Figure A.5. Medical needles and cells (includes seeds, tubes, and capsules) are primarily used for the interstitial or intercavitary treatment of tumors. Encapsulation is usually a platinum-iridium alloy or gold. Needles typically contain 1-10 mg of radium, are about 2 mm in diameter, and can range from about 1 to 6 cm in length. (Most are apparently in the range of 1-2 cm.) Cells typically contain 5-25 mg of radium and measure about 3 mm in diameter and 2 cm in length.

Plaques and nasopharyngeal applicators are generally designed to emphasize beta rather than gamma radiation. To achieve this, a less attenuating medium, such as monel metal, is used for encapsulation although some of the early plaques used a glazed or ceramic medium. Both of these types of sources are particularly susceptible to rupture and/or leakage due to poor construction, handling, or chemical or physical stresses. Plaques usually have activities in the range of 3 to 25 mg, and are used for the treatment of superficial skin lesions. Plaques typically are flat and measure 1-3 cm in diameter. A typical nasopharyngeal applicator contains 50 mg of radium and is mainly used for treatment of lymphoid tissue. Such applicators are typically contained in a 2 mm diameter by 2 cm long monel capsule attached to a 15-20 cm long handle.

Table A-21 provides a summary of the available data on medical radium sources as obtained from the EPA/HEW records, including activity and regional distributions. The top half of the table presents a series of eight activity ranges (in mg of radium), and the number of sources within each activity range is presented as a percentage. Also shown for each activity range is the average

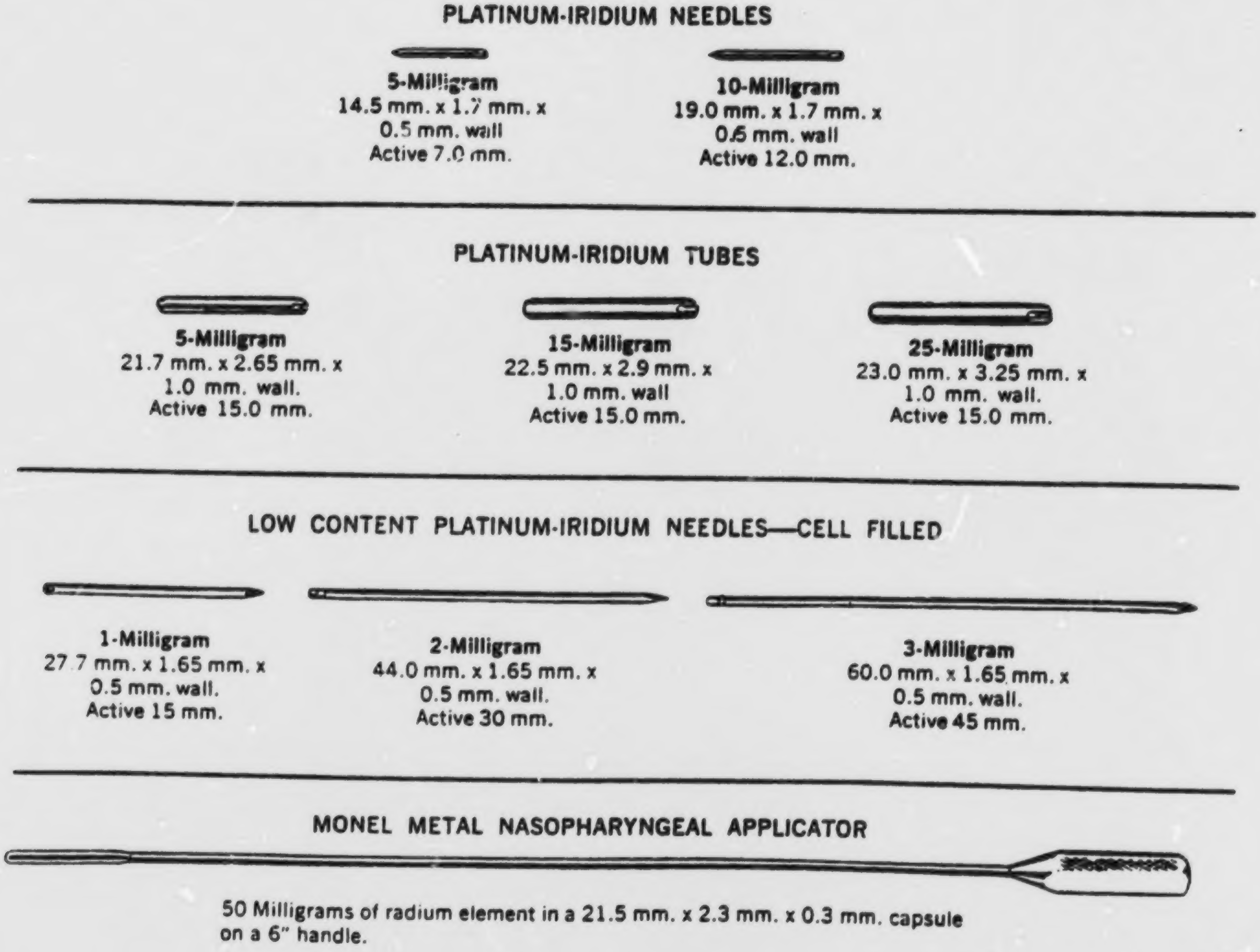
PLATINUM-IRIDIUM NEEDLES
5-Milligram
14.5 mm. x 1.7 mm. x
0.5 mm. wall
Active 7.0 mm.
10-Milligram
19.0 mm. x 1.7 mm. x
0.6 mm. wall
Active 12.0 mm.
PLATINUM-IRIDIUM TUBES
5-Milligram
21.7 mm. x 2.65 mm. x
1.0 mm. wall.
Active 15.0 mm.
15-Milligram
22.5 mm. x 2.9 mm. x
1.0 mm. wall
Active 15.0 mm.
25-Milligram
23.0 mm. x 3.25 mm. x
1.0 mm. wall.
Active 15.0 mm.
LOW CONTENT PLATINUM-IRIDIUM NEEDLES—CELL FILLED
1-Milligram
27.7 mm. x 1.65 mm. x
0.5 mm. wall.
Active 15 mm.
2-Milligram
44.0 mm. x 1.65 mm. x
0.5 mm. wall.
Active 30 mm.
3-Milligram
60.0 mm. x 1.65 mm. x
0.5 mm. wall.
Active 45 mm.
MONEL METAL NASOPHARYNGEAL APPLICATOR
50 Milligrams of radium element in a 21.5 mm. x 2.3 mm. x 0.3 mm. capsule on a 6" handle.

Figure A.5. Typical Medical Radium Sources

Table A-21. Summary of Data for Medical Radium Sources

	Needles	Cells	Plaques	Nasopharyngeal Applicators
Activity Range (% & ave. act. (mg)):				
≤ .1	0.01 (0.01)		0.6 (0.007)	
.1 ≤ .5	0.14 (0.45)	0.3 (0.46)		
.5 ≤ 1	2.2 (0.73)	5.4 (0.61)	0.6 (0.89)	
1 ≤ 5	52.1 (3.05)	40.9 (3.20)	30.6 (3.42)	3.1 (3.28)
5 ≤ 10	40.0 (7.72)	37.4 (7.57)	41.9 (8.18)	4.6 (9.44)
10 ≤ 50	5.3 (15.4)	15.8 (17.60)	24.1 (17.5)	81.5 (44.7)
50 ≤ 100	0.13 (71.7)	0.2 (66.9)	2.2 (61.0)	10.8 (67.4)
> 100	0.04 (118.)			
Weighted Ave. (mg)*	5.65	7.09	10.01	44.2
Total No. of Sources**	7,669	3,355	320	65
Regional Distribution (%):* **				
Region I	42.9	51.6	28.1	35.4
Region II	14.5	19.2	14.4	10.8
Region III	20.4	18.1	30.3	21.5
Region IV	13.2	7.6	17.2	18.5
Region V	9.1	3.4	10.0	13.8
Total No. of Sources**	7,668	3,355	320	65

*The weighted average activity is the average activity across all activity ranges for each medical waste stream.

**The total number of sources sometimes differs for calculation of activity range distribution and regional distribution. This is because some sources originated in Canada.

***The regional distribution across all medical sources is as follows: Region I: 45.0%, Region II: 15.8%, Region III: 20.0%, Region IV: 11.7%, Region V: 7.5% (total: 11,408 sources).

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activity of all the sources in each range. The mean activity--that is, the average activity across all eight activity ranges--is also shown. The largest number of data points is available for needles, while the fewest data points are available for nasopharyngeal applicators.

Radium has also been used in a very wide variety of non-medical applications, which increases the difficulty in characterizing non-medical waste radium sources. Based upon the data obtained from the HEW/EPA radium source collection program, it appears that two non-medical radium source waste streams can be identified: radium-beryllium sources and miscellaneous junk. Radium-beryllium sources exist in a wide activity range, with some exceeding 1000 mg of radium.

These sources are grouped together separately, since they include an additional radiation source--i.e., neutrons--to that emitted by almost all other forms of radioactive waste. The remaining non-medical sources are grouped together into a "miscellaneous" stream. Table A-22 lists the numerous descriptions of these various sources as obtained from the HEW/EPA records. A summary of the available EPA/HEW data on the activity and regional distribution of non-medical radium sources is presented in Table A-23. The format is essentially the same as that for Table A-21.

Projections

It is apparent that the physical and radiological characteristics of this type of waste dictate modifications in the standard approach used in this report to model impacts. The activity is highly concentrated in very small discrete sources, and some of the sources emit neutrons in addition to alpha, beta, and gamma radiation. Another factor is the difficulty in predicting the number of sources which may be available for disposal over the next few decades. Although the use of radium has greatly decreased over the last decade, it has been in commercial use since the turn of the century, and the last known documentation on the number of sources in use was in a HEW report which was published in 1971 and used 1968 data (Ref. 54). Another consideration is that something isn't waste until a decision is made to consider it so, and sources may easily be stored or used until an occurrence, such as leakage, leads the holder to desire to discard them.

Given the lack of any other data, a total of about 1100 radium sources is assumed to be available for disposal each year. This estimate is based upon the average receipt during the years 1974-1981 at the EPA Montgomery facility. These sources are assumed to be divided into the six waste streams discussed in the previous section according to ratios based upon the total number of sources identified in Tables A-21 and A-23. The waste streams, descriptions, and numbers of sources annually assumed to be generated in each waste stream are listed below.

Symbol	Description	Sources/year
N-RANEEDS	Medical needles	639
N-RACELLS	Medical cells	280
N-RAPLAQU	Medical plaques	27
N-RANPAPP	Medical nasopharyngeal applicators	5
N-RABESOR	Radium-beryllium neutron sources	7
N-RAMISCL	Miscellaneous non-medical sources	142

Table A-22. Example Descriptions of Non-Medical Radium Sources

Radium source
Radium capsule
Static eliminator bars
Radium D + E source
Screw plugs
Railroad density gauges
Radium liquid in bottles
Civil defense buttons
Plastic calibration sticks
Plastic calibration buttons
Steel cylinder
Cylinder in lead pig with gel
Capsule in plastic
Metal cylinder
Brass tube
Threaded brass caps
Glass cell
Glass ampule with liquid
Radium powder
Static eliminators in pipes
Plastic cubes
Vacuum gauge source
Luminous watch faces
Seeds
Radium disc
Penetron sources
Density gauge source in housing
Plated radium sources for alphatron vacuum gauge
Vial with yellow viscous liquid
Adhesive
Pellet
Radiography source
Radium dial powder
Platinum iridium rods
Liquid scintillation standards
Radium salt in glass tube
Calibration Source
Strip
Soil moisture gauge
Pipe wall thickness gauge
Radium check sources
Cone
Liquid depth gauge
Liquid level detectors
Diodes
Smoke detector sources
Foil disc
Alphatron vacuum gauge
"Ionatron" static eliminators
Chromatographic cells
Radium foils
Chromatographic foil

Table A-23. Summary of Data for Non-Medical Radium Sources

Ra-Be Sources

Activity Range (mg)	% & Ave. Act. (mg)
≤ 1	10.5 (0.36)
1 ≤ 5	14.0 (3.75)
5 ≤ 10	11.6 (9.49)
10 ≤ 50	14.0 (18.6)
50 ≤ 100	2.3 (100.)
100 ≤ 500	17.4 (302.)
500 ≤ 1000	4.7 (678.)
> 1,000	25.5 (1,431.5)

Weighted Ave. (mg) 457.

Total No. of Sources 86

Regional Distribution (%)	
Region I	25.4
Region II	27.0
Region III	11.1
Region IV	25.4
Region V	11.1

Total No. of Sources 63

Other Miscellaneous Sources

Activity Range	% & Ave. Act. (mg)
≤ .001	38.3 (6.13E-4)
.001 ≤ .01	36.0 (4.09E-3)
.01 ≤ 0.1	3.5 (4.87E-2)
0.1 ≤ 1	9.7 (4.88E-1)
1 ≤ 10	10.2 (3.17)
10 ≤ 100	0.4 (54.9)
100 ≤ 1,000	1.3 (464.)
> 1,000	0.5 (3,210.)

Weighted Ave. (mg) 22.7

Total No. of Sources 1,703

Regional Distribution (%)	
Region I	22.6
Region II	7.7
Region III	23.0
Region IV	46.1
Region V	0.6

Total No. of Sources 1,694

For the N-RANPAPP waste stream, each source is assumed to contain 50 mg of radium-226 in equilibrium with its initial short-lived daughters (i.e., through Radium C'). The data listed in Table A-21 contains some approximations, and it is believed that an insufficient number of data points are available to counteract these approximations. The activity across the remaining five waste streams is given as a distribution as shown in Table A-24. All four medical source waste streams are assumed to have the same regional distribution. Data for the regional distribution for each waste stream separately indicates similar trends. This distribution is as follows: Region I: 45.0%, Region II: 16%, Region III: 20.0%, Region IV: 12%, and Region V: 7%. The regional distribution of the N-RABESOR and N-RAMISCL waste streams is given below:

NRC Region	N-RABESOR	N-RAMISCL
I	25%	23%
II	27	8
III	11	23
IV	25	46
V	11	1

The chemical forms of the radium within the various sealed sources are difficult to project. However, one reference suggests that radium sources manufactured after 1940 are principally in the sulfate form. Few radium sources (less than 10%) still in circulation are projected to be in the more soluble chloride form (Ref. 56).

A.6.1.2 Radium Contaminated Ion-Exchange Resins

Data from the CRCPD suggests that ion-exchange resins contaminated with radium may be generated as part of a groundwater radium removal system currently being pilot tested by DOW Chemical (Ref. 55). The CRCPD estimates that this ion-exchange/filter media will have radium concentrations of about 1 mCi/ft^3 (35.31 mCi/m^3), and that a typical State may generate up to about 300 ft^3/yr (8.50 m^3/yr) of contaminated resin. If it is assumed that 25 States generate waste at this rate, this results in an annual volume of radium resins of about 7500 ft^3/yr (7.5 Ci/yr) (Ref. 55).

Using this data, a waste stream designated as N-RARESIN is assumed. This waste stream is assumed to have a concentration of 35 mCi/m^3 of radium-226 in equilibrium with all of its daughters. Since the DOW Chemical program referenced by the CRCPD is a pilot program, significant volumes of this waste stream would be expected only in the future. There is no information, however, to project when such significant volumes will be eventually generated. For this report, it is assumed to be in 1990. Starting in 1990, 7500 ft^3 (212.4 m^3) of resins are assumed to be generated in the United States each year. The regional distribution is assumed to be based on the human population distribution--i.e., 25% in Region I, 20% in Region II, 24% in Region III, 16% in Region IV, and 15% in Region V.

Table A-24. Activity Distributions Within Radium Source Waste Streams (mCi)

Activity Range (mCi)*	N-RANEEDS	N-RACELLS	N-RAPLAQU	N-RABESOR
≤ .1	0.01 (0.1)		0.6 (0.007)	
.1 ≤ .5	0.1 (0.45)	0.3 (0.46)		10.5 (0.36)
.5 ≤ 1	2.2 (0.73)	5.4 (0.61)	0.6 (0.89)	
1 ≤ 5	52.1 (3.05)	40.9 (3.20)	30.6 (3.42)	14.0 (3.75)
5 ≤ 10	40.0 (7.72)	37.4 (7.57)	41.9 (8.18)	11.6 (9.49)
10 ≤ 50	5.3 (15.4)	15.8 (17.6)	24.1 (17.5)	14.0 (18.6)
50 ≤ 100	0.1 (71.73)	0.2 (66.9)	2.2 (61.02)	2.3 (100.)
100 ≤ 500	0.04 (118.)			17.4 (302.)
500 ≤ 1,000				4.7 (678.)
> 1,000				25.5 (1,430.)
Weighted Ave. (mCi)	5.65	7.09	10.0	457.0

Activity Range (mCi)*	N-RAMISCL
≤ .001	38.3 (6.13E-4)
.001 ≤ .01	36.0 (4.09E-3)
.01 ≤ 0.1	3.5 (4.87E-2)
0.1 ≤ 1	9.7 (4.88E-1)
1 ≤ 10	10.2 (3.17)
10 ≤ 100	0.4 (54.9)
100 ≤ 1,000	1.3 (465.)
> 1,000	0.5 (3,210.)
Weighted Ave. (mCi)	22.7

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*The first number is the percentage of generated waste sources assumed to be within each listed activity range. The second number, in parentheses, is the average activity in each range.

A.6.2 Military Waste

Projections of military waste are made based upon a review of low-level radioactive waste shipment records for the year 1982. The review indicated that about 29,600 ft^3 of waste from military sources was disposed during 1982, not including waste generated from military-oriented institutional facilities such as Veterans Administration hospitals, Walter Reed Army Medical Center, and so forth. (Such waste is considered as part of other institutional waste sources in Section A.4.4). The volume distribution of this military waste was as follows:

Service	Volume (ft^3)	Volume %
Unidentified	442.5	1.5
Marines	85.8	0.3
Navy	15,432.2	51.5
Air Force	1,444.3	4.8
Army	12,987.9	42.0
Total:	29,992.7	

As shown, about half of the waste volume was generated by the Navy.

The records indicate that the waste thus produced is a cross section of waste produced by other sources--i.e., waste from nuclear reactor operations, tritium contaminated waste, laboratory waste, source material, activated metal, sealed sources, etc. In general, Army waste was dominated by source material--generally depleted uranium. Navy waste was dominated by waste from nuclear reactor operations (e.g., submarines). Waste contaminated with large quantities of tritium was generated from Army, Navy and Air Force sources.

In this report, military wastes containing large quantities of tritium have already been considered in Section A.5.2.4. Similarly, waste containing source material appears to be very similar to that generated by industrial sources (N-SSWASTE and N-SSTRASH) considered in Section A.5.1. Waste containing Ra-226 is considered elsewhere in this chapter. This leaves waste from Naval reactor operations, which in any case is the largest waste source in terms of volume.

In 1982, waste from Naval reactor operations originated from six sites, and totaled 14,451 ft^3 and 49,535 mCi. Of this waste, about 12% originated in NRC Region I, 54% in NRC Region II, and 34% in NRC Region V. The records indicate that the waste was dominated by activation products such as Co-60. The records are a bit obscure, but apparently about 87% of this waste consists of "dry waste," including dry compressible waste, contaminated equipment, and so forth. The remaining 13% of this waste can be characterized as "wet waste," including spent resins, filter sludges, solidified liquids, etc. One reason that some uncertainties exist is that frequently a package containing wet wastes (solidified liquids and ion-exchange resins) also contained a certain fraction of dry wastes such as trash.

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Based upon this data, two Naval operations waste streams are projected: M-NAVYDRY and M-NAVYWET, where "M" refers to "military." The M-NAVYDRY waste stream consists of dry compressible material, contaminated equipment, and so forth. The data is judged to be insufficient to justify breaking the waste stream further into compactible and non-compactible waste streams. Based upon the records, the average gross concentration is projected to be approximately 2.00E-2 Ci/m^3. The M-NAVYWET waste stream is a mixture of dewatered resins and filter media, solidified resins and filter media, and solidified liquid. Again, the data is judged to be insufficient to justify breaking the waste stream into smaller components. The records do indicate, however, that most of the wet waste (about 80%) appears to be solidified in cement.

Insufficient data is available to project any decreases or increases in volume over the next 20-30 years. In any case, one would expect that the waste volume would be geared to considerations such as the size of the nuclear Navy, which involves policy decisions which are difficult to project at this time. Thus, the same volume is projected to be generated each year for both waste streams. The projected volumes and gross concentrations are as follows:

Waste Stream	Volume/Year (m^3/yr)	Gross Conc. (Ci/m^3)
M-NAVYDRY	354	0.02
M-NAVYWET	55	0.79

The M-NAVYWET waste stream is given as a distribution. The records indicate that about 94% of the M-NAVYWET waste stream is of relatively low concentration (an average of about 0.2 Ci/m^3) while about 6% of the waste stream is in much higher concentrations (an average of about 10 Ci/m^3). This gives a weighted average gross concentration of about 0.788 Ci/m^3.

The regional distribution for the two waste streams is projected to be the same as that given above.

A.6.3 Future Potential Waste Generation

Future generation of non-fuel cycle waste may arise from a number of sources. Examples could include decontamination and decommissioning of facilities such as those for industrial manufacturers of sealed sources; manufacture of tritiated products and radiopharmaceuticals; and research, education, and test reactors. Generation of such waste is currently difficult to project, although it is believed that wastes that exceed Class C concentrations will generally consist of transuranic-contaminated scrap and trash. This waste will be generated from eventual decommissioning of sealed source manufacturers using isotopes such as Am-241 or Pu-238. Small quantities of activated metals, and even smaller quantities of sealed sources, may also be generated.

One sealed source manufacturer, located in NRC Region III, which at one time had one of the largest Am-241 operations in the country, has recently closed down the facility and is in the process of decommissioning it. At one time as much as 160 m^3 of waste exceeding Class C concentrations was conservatively projected to be generated from decommissioning the facility (Ref. 57). Current projections based on ongoing experience are on the order of 60 m^3 (Ref. 58). Another closed facility located in NRC Region I, which was a former manufacturer of smoke detector sources, may also generate decommissioning wastes exceeding Class C concentrations. A conservative estimate is on the order of an additional 60 m^3 (Ref. 58).

Currently operating sealed source manufacturers may not generate nearly as much waste during decommissioning, simply due to the greater use of recycled materials during operations. (One reason that the above facility closed operations may be that the operators were incapable of performing extensive recycle of radioactive material, while other operators of manufacturing facilities appear to recycle extensively.) Waste generation will probably depend less upon the throughput of material through the facility, and more upon the facility design (e.g., whether there are traps within the manufacturing facility which collect material and impede decontamination) and operating philosophy.

One reference estimates that eventual decommissioning of transuranic sealed source manufacturing facilities in current operation may generate up to 230 m^3 of waste that exceeds Class C concentrations (Ref. 58). These estimates are very rough and the authors are aware of no plans by the existing manufacturers to cease operations. These estimates, however, do imply that such waste generation would be expected to be comparatively small.

A.7 NON-ROUTINE WASTE SOURCES

This includes a significant number of waste streams which are considered to be hypothetical, unusual, or will be generated far into the future. This includes waste from uranium fuel recycle, nuclear power plant decommissioning, and the West Valley Demonstration Project. Waste from decontamination of the Three Mile Island, Unit 2, facility is also briefly considered.

A.7.1 Uranium Fuel Recycle

A.7.1.1 Waste Generation Overview

The following description of the reference uranium fuel cycle facilities is taken from reference 59.

Fuel Reprocessing Facility

The reference fuel reprocessing facility is based upon the uncompleted facility constructed by Allied-General Nuclear Services (AGNS) near Barnwell, South Carolina. This facility is assumed to have a fuel processing capacity of 2000 MTHM/yr (MTHM: metric tons of heavy metal). Spent fuel having an average burnup of 29,000 MWd/MTHM (megawatt-days per metric ton of heavy metal) is assumed to be aged at least 6 months after discharge from a reactor prior to receipt at the reprocessing facility; it is further stored at the reprocessing facility for one year prior to processing. These are conservative assumptions. If fuel reprocessing ever became a reality, it would take several years to work off the existing backlog of spent fuel, much of which has already aged for several years.

A simplified flow diagram of the reprocessing operations is provided as Figure A.6, which also illustrates the sources of much of the liquid and solid wastes that result from plant operation (Ref. 59). The overall operation of

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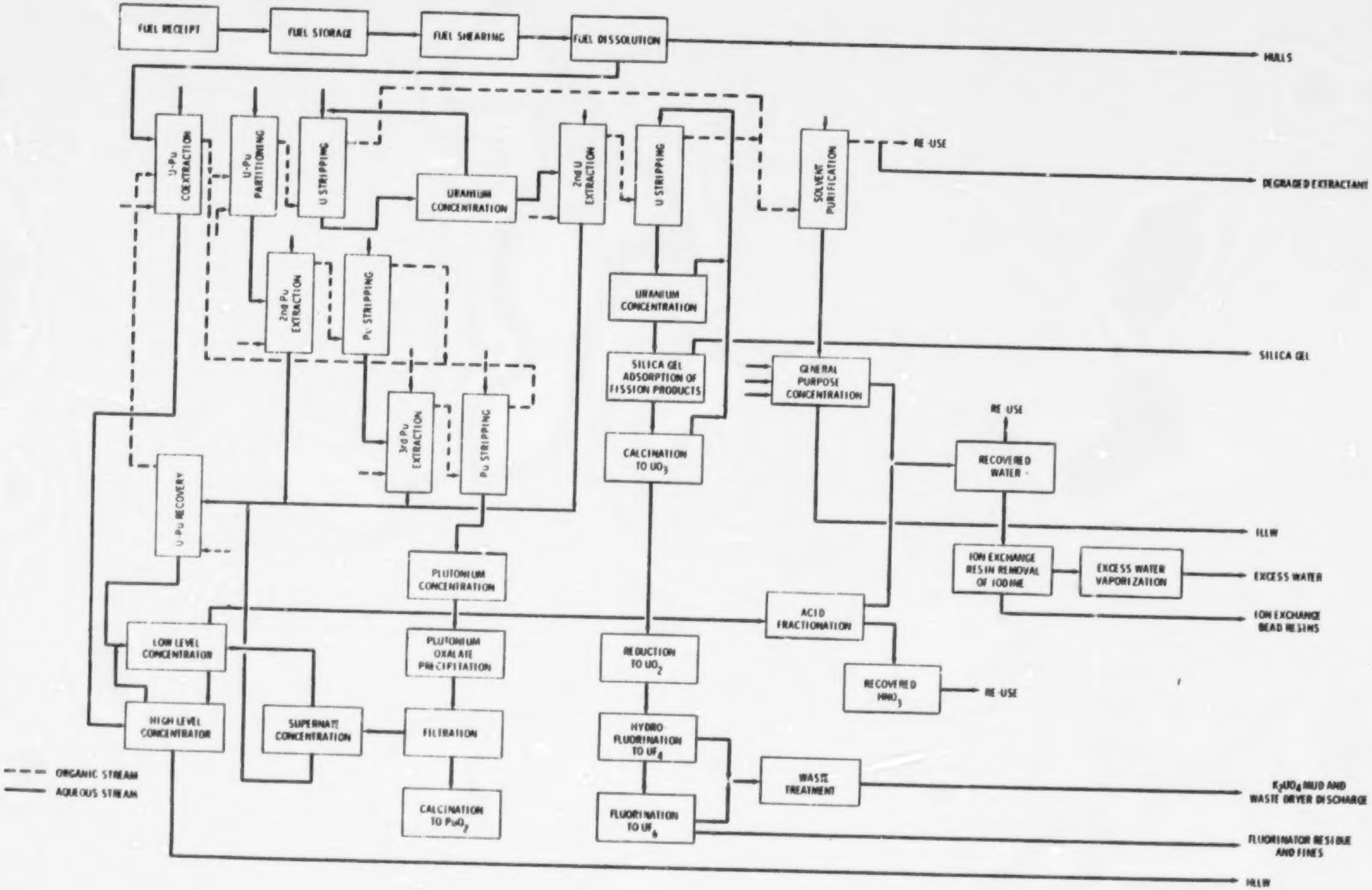
FUEL RECEIPT
FUEL STORAGE
FUEL SHEARING
FUEL DISSOLUTION
HULLS
U-Pu COEXTRACTION
U-Pu PARTITIONING
U STRIPPING
URANIUM CONCENTRATION
2nd U EXTRACTION
U STRIPPING
SOLVENT PURIFICATION
RE-USE
DEGRADED EXTRACTANT
2nd Pu EXTRACTION
Pu STRIPPING
URANIUM CONCENTRATION
SILICA GEL ADSORPTION OF FISSION PRODUCTS
GENERAL PURPOSE CONCENTRATION
SILICA GEL
3rd Pu EXTRACTION
Pu STRIPPING
CALCINATION TO UO_3
RE-USE
RECOVERED WATER
U-Pu RECOVERY
HLLW
PLUTONIUM CONCENTRATION
ION EXCHANGE RESIN REMOVAL OF IODINE
EXCESS WATER VAPORIZATION
EXCESS WATER
ACID FRACTIONATION
ION EXCHANGE BEAD RESINS
LOW LEVEL CONCENTRATOR
PLUTONIUM OXALATE PRECIPITATION
REDUCTION TO UO_2
RECOVERED HNO_3
RE-USE
HIGH LEVEL CONCENTRATOR
SUPERNATE CONCENTRATION
FILTRATION
HYDROFLUORINATION TO UF_4
WASTE TREATMENT
K_2UO_4 MUD AND WASTE DRYER DISCHARGE
ORGANIC STREAM
AQUEOUS STREAM
CALCINATION TO PuO_2
FLUORINATION TO UF_6
FLUORINATOR RESIDUE AND FINES
HLLW

Figure A.6. Flow Diagram of Reprocessing Operations

the facility can also be subdivided into four separate suboperations. These include fuel receiving and storage, main plant activities, plutonium conversion, and uranium conversion.

Spent fuel arrives at the facility in shielded casks and is stored in a water basin at the fuel receiving and storage facility. Storage operations are carried out in a generally similar manner to fuel storage operations at a nuclear power plant, and similar waste streams are generated. These waste streams include ion-exchange resins, filter sludge, and concentrated and other liquids from basin water cleanup operations, as well as trash.

The next step is fuel shearing. Fuel elements are chopped into short segments a few centimeters in length and are then dropped into a vat containing hot nitric acid. The nitric acid dissolves the residual fuel, producing an aqueous nitrate solution containing the great majority of the residual uranium and plutonium as well as the fission products. The undissolved fuel cladding, or "hulls," is removed as a solid waste stream.

A series of solvent extraction operations using tributyl phosphate (TBP) as a hydrocarbon dilutent is then performed to separate the uranium and plutonium from each other and from the fission products. The first extraction process results in separation of greater than 99% of the fission products as an aqueous solution; this solution is then concentrated to form high-level liquid waste. Some (about 0.5%) of the uranium and plutonium is also carried with the high level waste stream.*

Separation of the uranium from the plutonium is carried out in subsequent solvent extraction cycles. The separate constituents are further purified, which for uranium includes removal of residual fission products using solvent extraction and absorption of fission products on silica gel. The silica gel eventually becomes a waste stream. Plutonium is further purified using additional solvent extraction processes. Aqueous liquids resulting from these various purification processes are combined and concentrated, and are termed intermediate level liquid waste. During the process, nitric acid is recovered for reuse. Also recovered and purified for reuse is the TBP. The TBP, however, eventually becomes too degraded for use and eventually becomes a waste stream.

Product streams are in the form of aqueous solutions--i.e., as uranyl nitrate and plutonium nitrate. The plutonium nitrate is converted to a solid powder by addition of oxalic acid. This precipitates into plutonium oxalate, which is subsequently filtered off and calcined to form plutonium dioxide (PuO_2).

Uranyl nitrate is converted to uranium hexafluoride (UF_6), which is a solid at room temperature but sublimates when sufficiently heated. In the process, the uranyl nitrate is first calcined to uranium trioxide (UO_3), which is reduced to uranium dioxide (UO_2) using cracked ammonia. The UO_2 is converted to uranium tetrafluoride by addition of hydrogen fluoride, and then converted to UF_6 by reaction with fluorine in a fluidized bed.

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*This waste stream would in reality be solidified in a media such as borosilicate glass and eventually disposed in a high-level waste repository. For the purposes of this report, however, the waste stream is assumed to be handled like any other liquid generated as part of operations. This is done solely to provide a means to determine a rough level of relative impacts.

The uranium conversion process generates considerably more waste than the plutonium conversion process. One waste stream which is expected to contain some transuranic contamination consists of the bed residues and fines generated as part of operation of the fluidized bed fluorinator. Other waste streams are generated as a result of treatment of the airborne effluents resulting from the conversion process. One stream consists of K_2UO_4 mud, which is the insoluble residue remaining after KOH and KF are decanted from an off-gas cleanup solution. The largest waste stream results from recovery of the KOH for reuse in the process. This results from treating the KOH-KF solution with lime to precipitate calcium fluoride (CaF_2) and recover KOH. The CaF_2 waste stream is filtered and dried to become UF_6 plant waste dryer discharge. This waste stream is very low in activity but large in volume.

Other waste streams generated at the reprocessing facility include ion exchange resins, combustible and noncombustible trash, high efficiency particulate air (HEPA) filters from air treatment processes, and failed equipment. The ion exchange resins are generated as part of miscellaneous water cleanup activities associated with main plant operations. The other waste streams (trash, HEPA filters, failed equipment) are generated at all process suboperations, although the activity levels and radionuclide distributions will vary depending upon the operation. For example, failed equipment generated as part of main plant operations may be highly contaminated with fission products and actinides, while failed equipment generated as part of uranium conversion would only be slightly contaminated with uranium.

Mixed Oxide (MOX) Plant

The reference MOX fuel fabrication plant is based upon proven, currently available technology, and has a throughput of 400 MTHM/yr, which approximately matches the output of the reference fuel reprocessing plant. A 1-year delay is assumed between plutonium recovery at the reprocessing plant and fabrication of MOX fuel.

An illustration of the major processing operations is provided as Figure A.7, which also indicates sources of liquid wastes. The MOX plant receives UO_2 and PuO_2 powder from off-site suppliers, and mechanically blends these powders into appropriate proportions (Ref. 59). The mixed oxides are pressed into cakes, granulated and reblended, and then pressed into pellets. The pellets are then sintered, finish ground, washed, inspected, and loaded into fuel rods (zircaloy tubing). The fuel rods are completed and sealed, and then go through a series of finishing and inspection processes such as degreasing, etching and rinsing, autoclaving and leak testing. The finished fuel rods may then be packaged and shipped to a uranium fuel fabrication facility for inclusion with enriched uranium fuel rods in fuel assemblies.

Waste generated from process operations include wet wastes, combustible and compactible wastes, and noncompactible, noncombustible wastes such as failed equipment. Wet wastes include process solutions from fuel pellet washing, fuel rod etching, and decontamination operations. Other liquid wastes consist of concentrated liquids generated as part of a dirty scrap recovery process. Combustible and compactible wastes include paper, plastic, disposable protective clothing, and discarded HEPA filters from airborne effluent treatment systems. Noncompactible, noncombustible wastes include scrap zircaloy tubing, discarded equipment, glove box enclosures, glass, and so forth.

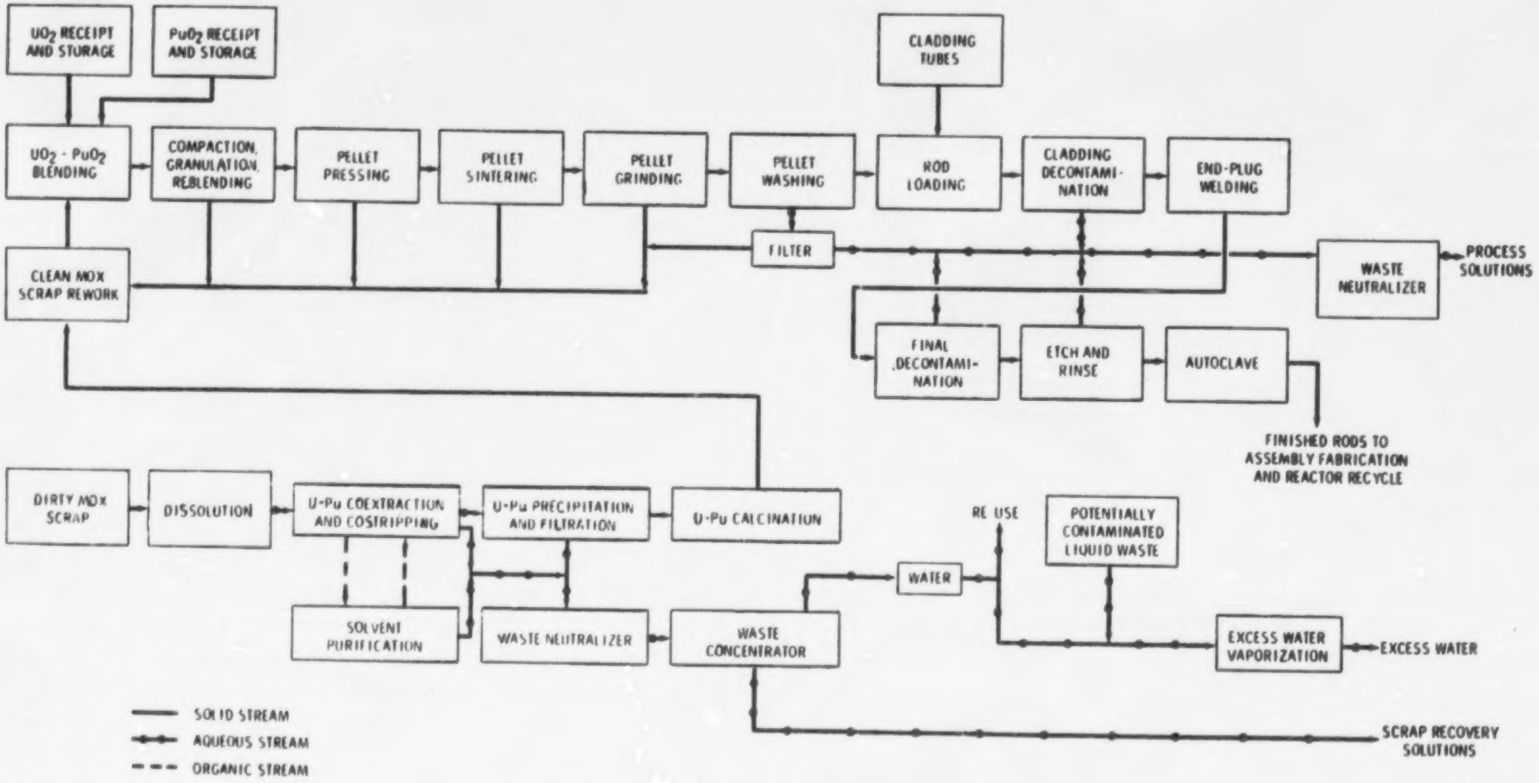
UO$_2$ RECEIPT AND STORAGE
PuO$_2$ RECEIPT AND STORAGE
CLADDING TUBES
UO$_2$ - PuO$_2$ BLENDING
COMPACTION, GRANULATION, REBLENDING
PELLET PRESSING
PELLET SINTERING
PELLET GRINDING
PELLET WASHING
ROD LOADING
CLADDING DECONTAMINATION
END-PLUG WELDING
FILTER
CLEAN MOX SCRAP REWORK
WASTE NEUTRALIZER
PROCESS SOLUTIONS
FINAL DECONTAMINATION
ETCH AND RINSE
AUTOCLAVE
FINISHED RODS TO ASSEMBLY FABRICATION AND REACTOR RECYCLE
DIRTY MOX SCRAP
DISSOLUTION
U-Pu COEXTRACTION AND COSTRIPPING
U-Pu PRECIPITATION AND FILTRATION
U-Pu CALCINATION
RE USE
POTENTIALLY CONTAMINATED LIQUID WASTE
WATER
SOLVENT PURIFICATION
WASTE NEUTRALIZER
WASTE CONCENTRATOR
EXCESS WATER VAPORIZATION
EXCESS WATER
SCRAP RECOVERY SOLUTIONS
SOLID STREAM
AQUEOUS STREAM
ORGANIC STREAM

Figure A.7. Major Processing Operations at MOX Fuel Fabrication Facility

A.7.1.2 Waste Stream Projections

Waste streams projected to result from uranium fuel recycle are projected based upon information presented in references 4 and 59. It is obvious that the future generation of such waste is speculative. To analyze potential impacts, however, estimated waste streams and volumes are presented for a single reference 2000 MTHM/yr nuclear fuel reprocessing plant and a single reference 400 MTHM/yr mixed oxide (MOX) fuel fabrication plant. Such estimates are made assuming that both facilities are operated at maximum capacity. Impacts are then calculated by assuming that a given number of facilities are operated within a given region. The number of facilities, year in which the facilities start operating, and region(s) in which the facilities operate are options which may be varied by the user of the calculational methodology.

Using references 4 and 59, 37 separate waste streams can be identified from the two uranium fuel recycle facilities. These are reduced to a somewhat more manageable number (23) by combining some waste streams having similar physical and radiological characteristics. The waste streams, a brief description of them, and the projected annual volumes are listed in Table A-25. Note that the prefix "R" (for "recycle") is used to designate this group of waste streams. Also note that the wastes from the nuclear fuel reprocessing plant can be separated according to their origin: the main plant (MP), the fuel storage basin (SB), the uranium hexafluoride (UF_6) conversion process (UF), and the plutonium dioxide (PuO_2) conversion process (PU).

Total waste volumes are projected to be 8949 m^3/yr from the fuel reprocessing plant and 468 m^2/yr from the MOX facility. Volumes are given in as-generated forms, and future waste processing and packaging procedures will somewhat modify the volumes and corresponding radionuclide concentrations, as well as the waste densities.

A.7.2 Nuclear Power Plant Decommissioning

Nuclear fuel cycle facilities will eventually reach the end of their useful lives and would then be considered candidates for decontamination and decommissioning. Four general types of nuclear fuel cycle facilities can be considered: uranium fuel fabrication plants, uranium fuel recycle facilities, research and test facilities, and light water reactors. Decommissioning of nuclear fuel fabrication plants would be expected to generate a relatively insignificant volume of waste requiring offsite shipment. Since this waste would be principally contaminated with uranium isotopes, it would all be considered under 10 CFR Part 61 to be Class A. The timing of such decommissioning is also speculative, and would probably depend on economic rather than safety considerations. Decommissioning of uranium fuel recycle facilities would generate waste having a wide spectrum of activities. However, such facilities would have to be constructed and operated first, and so generation of the waste would be well removed in time (e.g., several decades). Additional information is provided in the draft Part 61 EIS (Ref. 3).

Relatively small quantities of wastes are projected to result from eventual decommissioning of facilities performing destructive radiochemical inspection of irradiated nuclear reactor fuels. (Operation at the three existing facilities has been described in Section A.2.2.3.) The authors are not aware of any

Table A-25. Nuclear Fuel Recycle Waste Streams and Volumes

Waste Stream	Description	Annual Volume (m^3/yr)	Concentration (Ci/m^3)
Fuel Reprocessing Plant			
· Main Plant			
R-HLLWFRP	High level liquid waste	1200	2.36E+6
R-FUEHARD	Fuel assembly hardware	112	1.49E+5
R-HULLFRP	Hulls from chop/leach process	532	1.66E+3
R-ILLWFRP	Intermediate level liquid waste	220	1.68E+3
R-SILIGEL	Silica gel	10	3.73E+0
R-MPCOTRH	High activity compressible trash plus ventilation filters	803	9.78E-1
R-MPCOTRL	Low activity compressible trash	2400	2.99E-4
R-MPNCTRA	Noncompressible trash plus failed equipment	1200	6.01E-1
R-DEGREXT	Degraded extractant (TBP)	16	2.38E+3
R-MPRESIN	Ion exchange resins	10	3.77E+2
· Storage Basin			
R-SBRESIN	Ion exchange resins plus filter sludge	18	5.02E+1
R-SBCOLIQ	Concentrated liquids (sulfate concentrate plus misc. solutions)	30	4.28E+1
R-SBCOTRA	Compressible trash plus ventilation filters	1220	3.44E-2
R-SBNCTRA	Noncompressible trash plus failed equipment	120	1.50E-2
· UF_6 Conversion			
R-UFFINES	Fluorinator residues plus fines	68	6.63E+0
R-UFK2MUD	K_2UO_4 mud	140	1.73E-2
R-UFCOTRA	Compressible trash plus ventilation filters	110	8.68E-3
R-UFNCTRA	Noncompressible trash plus failed equipment	40	5.15E-3
· PuO_2 Conversion			
R-PUCOTRA	Compressible trash plus ventilation filters	100	8.81E+3
R-PUNCTRA	Noncompressible trash plus failed equipment	52	1.35E+3
Mixed Oxide Facility			
R-MOXCOTR	Compressible trash plus ventilation filters	240	1.01E+3
R-MOXNCTR	Noncompressible trash plus failed equipment	160	1.66E+2
R-MOXSOLN	Process plus scrap recovery solutions	148	2.35E+2

plans to decommission any of the three facilities performing the fuel inspections, nor of any technical reason which would limit their useful life. The three facilities perform these studies in shielded hot cells, which can be envisioned as national technical resources, and perform work having both commercial and governmental applications.

Some very rough estimates of eventual decommissioning waste volumes have been made in reference 37. This reference estimates that eventual decommissioning volumes will be on the order of 285 m^3 of waste per facility, of which only about half would have concentrations that exceed Class C concentrations. This does not include the small volumes of operational wastes which are currently being stored.

The most significant source of decommissioning waste will arise from decommissioning light water reactors. It is difficult to project the resulting waste volumes and activities, since the generation of decommissioning waste will depend upon a number of factors such as the length of service life of a plant prior to decommissioning, the size and design of a plant, the operating history of the plant, and the decommissioning mode undertaken (e.g., immediate dismantle after shutdown vs. deferring dismantlement for up to several years following shutdown). Although a number of studies have been made of the volumes and activities expected from decommissioning, the volumes and activities are generally estimated for large modern units and such units are not expected to undergo decommissioning until well after the year 2000. Reactors potentially dismantled prior to the year 2000 are expected to be considerably smaller in capacity, have shorter operating lives than the reactors generally used as models for the studies and are expected to generate considerably lower waste volumes and/or activities. Another important consideration is the existence, or lack of existence, of Federal storage or disposal facilities for spent fuel.

The types of wastes that will be generated during decommissioning will probably be very similar to those generated during normal operations, but the relative quantities of the different waste streams may differ. Radionuclide distributions may also differ significantly. Although one would generally expect that waste volumes would be lower for a smaller sized plant, it is not clear that the activity would be lower. One study (Ref. 60), for example, postulates that activated metal waste may be higher in specific activity at a smaller (550 MW(e)) plant than at a larger PWR plant (1160 MW(e)), due to higher neutron fluxes in the smaller reactor vessel.

Another consideration is that reactors may be shut down and dismantled for policy and economic reasons rather than purely technical reasons. There are a number of early low power units generally constructed as demonstration projects forerunning larger, more economical to operate units with capacities on the order of several hundred to a thousand MW(e). Although utilities would generally prefer to keep the older units operable for as long as they are cost-effective, costs of upgrading the older units to meet new NRC safety requirements may result in some of the older plants being decommissioned prior to the end of their otherwise serviceable lives.

Thus, projections of decommissioning volumes and activities involve three different activities: (1) generating a manageable number of separate waste streams, (2) determining a method to proportion the waste generation rate to determinable characteristics such as plant size and/or operating life, and (3) approximating when particular nuclear power plants are likely to undergo decommissioning. Any of these three activities involves several uncertainties.

The approach taken is to split the first two considerations from the third one. First, some background information is presented from several sources in the literature. This provides the reader some input on different possible approaches to correlate waste streams, volumes, and activities to quantifiable measures such as power ratings and operating lifetimes. This background section is then followed by sections summarizing the approaches used in this report for estimating waste streams, volumes, and activity distributions. Following this is a section discussing the assumptions used for assessing the timing of power plant decommissioning.

A.7.2.1 Background

Some of the more recent work on estimating waste volumes and activities from power plant decommissioning has been performed by Pacific Northwest Laboratories (PNL). Two studies have been published which analyze the technology, safety, and cost of decommissioning a large reference PWR (Ref. 61) and a large reference BWR (Ref. 62). The model for the reference PWR is the Portland General Electric Company Trojan Nuclear Plant having a generating capacity of 1175 MW(e) (3500 MW(t)), and using a Westinghouse four-loop nuclear steam supply system. The model for the reference BWR is the Washington Nuclear Project No. 2 (WNP-2) at Hanford, Washington. This 1155 MW(e) (3320 MW(t)) unit, which started operation in 1984, uses a General Electric BWR-5 nuclear steam supply system. The plant uses a Mark-II containment.

A summary of waste volumes and activities for the two reference LWRs is provided in Table A-26. These estimates are based on information in references 1, 63, and 64. The volumes and activities are projected from an assumption of immediate dismantlement of the plant following 40 calendar years of operation at 75% of full power, or 30 effective full power years (EFPY). Dismantlement of the reference PWR is projected to require 4 years, while dismantlement of the reference BWR is projected to require 3-1/2 years.

The volumes and activities summarized in Table A-26 should be interpreted with some care. Actual volumes and activities from decommissioning a given LWR may be highly site specific and a function of such factors as the size and design of the unit, the rated power level, the amount of time spent at full power, and the time between shutdown and dismantlement. However, it is apparent that on the order of 99% of the activity from decommissioning wastes will be contained in the activated metal. Relative volumes and activities for various activated metal components are shown in Table A-27. Based upon data obtained from the PNL references, the activated metal components are estimated to be principally composed of either stainless or carbon steel. As shown, specific activities of BWR activated components are estimated to vary by four orders of magnitude, while PWR components by six orders of magnitude. Of special interest for disposal purposes are the BWR core shroud and the PWR core shroud and lower grid plate.

The PNL studies also considered the impacts on the projected waste volumes of different sized reactor plants--all, however, operating for the same amount of time. The information on different sized PWRs is contained in reference 65, while the information on BWRs is contained in reference 62. The approach taken for both types of plants is to briefly examine different sized plants, estimate

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Table A-26. Summary of Wastes from Decommissioning a Reference PWR and a Reference BWR

Waste Stream	Volume (m^3)	Activity (Ci)
Reference 1155 MW(e) BWR:		
Activated metal#	138	6,552,310
Activated concrete#	90	170
Contaminated metal#	14,629	8,400
Contaminated concrete#	2,600	144
Dry solid waste (trash)*	3,390	1,806
Spent resins**	42	228
Filter cartridges†	--	--
Evaporator bottoms††	438	32,753
Reference 1175 MW(e) PWR:		
Activated metal#	485	4,840,820
Activated concrete#	707	2,000
Contaminated metal#	5,465	900
Contaminated concrete#	10,613	100
Dry solid waste (trash)*	1,415	757
Spent resins**	30	42,000
Filter cartridges†	8.9	5,000
Evaporator bottoms††	133	13,805

#As-packaged volumes.

*Volumes are shown as-generated (prior to additional treatment such as compaction or incineration). Most of the trash is considered to be combustible.

**BWR spent resins actually include spent resins and filter sludge. Volumes shown are dewatered volumes.

†PWR filter cartridge volumes are given as-solidified in concrete in 55-gallon drums. Filter cartridges are assumed not to be used in the BWR wet waste treatment system.

††PWR and BWR evaporator bottom volumes are given as-generated prior to solidification. An assumed volume reduction factor of 1.4 was used to convert solidified (but unpackaged) volumes in references 63 and 64 to an unsolidified form.

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Table A-27. Volumes and Activities of Decommissioned LWR Activated Metals

Component	Const.** Mat.	Disposal Volume* (m³)	Activity (Ci)	Disposal Concentration (Ci/m³)
Reference BWR:				
Steam separator assembly	S	10	9,600	960
Fuel support pieces	S	5	700	140
Control rods and in-core instruments	S	15	189,000	12,600
Control rod guide tubes	S	4	100	25
Jet pump assemblies	S	14	20,000	1,429
Top fuel guide	S	24	30,100	1,254
Core support plate	S	11	650	59
Core shroud	S	47	6,300,000	134,043
Reactor vessel wall	C	8	2,160	270
Total		138	6,552,310	
Reference PWR:				
Pressure vessel cylindrical wall	C	108	19,170	178
Vessel head	C	57	<10	0.18
Vessel bottom	C	57	<10	0.18
Upper core support assembly	S	11	<10	0.91
Upper support columns	S	11	<100	9.1
Upper core barrel	S	6	<1,000	167
Upper core grid plate	S	14	24,310	1,736
Guide tubes	S	17	<100	6
Lower core barrel	S	91	651,000	7,154
Thermal shields	S	17	146,100	8,594
Core shroud	S	11	3,431,100	311,909
Lower grid plate	S	14	553,400	39,529
Lower support columns	S	3	10,000	333
Lower core forging	S	31	2,500	81
Miscellaneous internals	S	23	2,000	87
Reactor cavity liner	S	14	<10	0.7
Total		485	4,840,820	

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*Disposal volumes include the disposal container after the activated metal components have been cut into manageable pieces.

**Construction materials. S: stainless steel, C: carbon steel

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the quantity of waste that would be generated for certain major components (either in terms of mass or volume), and then ratio the waste quantities to that estimated for the reference plant. Curve fits are then made to the scaling factors for the individual and combined components to determine waste quantities (and ultimately costs) as a function of reactor capacity.

Including the reference PWR, four PWRs were examined by PNL, including one small and three large plants. The thermal power ratings and scaling factors calculated for the various components are summarized in Table A-28 (Ref. 65). For example, the scaling factor 0.848 for the Turkey Point reactor vessel merely means that the Turkey Point vessel is 0.848 the mass of the Trojan vessel.

Table A-28. Individual PWR Component Scaling Factors for Immediate Dismantlement

Component	Yankee-Rowe (600 MW(t))	R. E. Ginna (1,300 MW(t))	Turkey Point (2,550 MW(t))	Trojan (3,500 MW(t))
Reactor	0.343	0.548	0.848	1.0
Vessel internals	0.180	0.665	0.867	1.0
Pumps and piping	0.547	0.493	0.757	1.0
Steam generators	0.289	0.424	0.635	1.0
Pressurizer	0.172	0.316	0.684	1.0
Containment bldg.	0.290	0.379	0.671	1.0
Weighted total	0.288	0.521	0.782	1.0

The component headings "reactor vessel" and "vessel internals" are self explanatory, as are the terms "steam generators" and "pressurizer." "Pumps and piping" refers to the primary coolant pumps and coolant piping. The component "containment building" includes various types of waste such as contaminated and activated concrete, contaminated metal, and small amounts of activated metal. The activated metal and concrete primarily comes from material forming the biological shield within the reactor cavity. This material will also probably be slightly contaminated.

Based on this data, PNL then curve fitted the individual component scaling factors to facilitate interpolation. The data sets are fitted to equations of the form $SF_i = A_i + B_i\ (MWt) + C_i\ (MWt)^2 + D_i\ (MWt)^3$, where the subscript i refers to the ith major component. In two cases (for vessel internals and pumps and piping), PNL does not use the Yankee-Rowe scaling factors because they depart significantly from the general pattern for the other three plants. The derived fitted curves are as follows:

Component	Fitted Scaling Factors
Reactor vessel	0.130 + 3.589E-4 (MWt) - 2.713E-8 (MWt)^2 - 1.258E-12 (MWt)^3
Vessel internals	0.400 + 2.318E-4 (MWt) - 2.397E-8 (MWt)^2 + 1.925E-12 (MWt)^3
Pumps and piping	0.320 + 8.346E-5 (MWt) + 4.201E-8 (MWt)^2 - 2.955E-12 (MWt)^3
Steam generators	0.200 + 1.270E-4 (MWt) + 1.713E-8 (MWt)^2 + 2.696E-12 (MWt)^3
Pressurizer	0.070 + 1.591E-4 (MWt) + 2.267E-8 (MWt)^2 + 2.230E-12 (MWt)^3
Containment bldg.	0.230 + 5.829E-5 (MWt) + 4.163E-8 (MWt)^2 + 1.307E-12 (MWt)^3

A similar approach was taken for BWRs (Ref. 62). In this case, seven similar sized plants were examined including the reference plant (WNP-2). A summary of the plant power ratings and scaling factors is presented in Table A-29.

Table A-29. Plant-Specific Scaling Factors for Selected BWR Power Plants

		Component - Specific Scaling Factors		
Power Plant	Power Rating (MWt)	Reactor Vessel	Vessel Internals	Reactor Building
Vermont Yankee	1,593	0.572	0.531	0.611
Oyster Creek	1,600	0.791 (a)	0.561	0.620
Monticello	1,670	0.588	0.617	0.612
Cooper	2,381	0.757	0.705	0.748
Dresden 2 or 3	2,527	0.782	0.731	0.826
Peach Bottom 2 or 3	3,293	0.910	0.734	0.995
WNP-2	3,320	1.000	1.000	1.000
Cost factors:		0.0764	0.3983	0.5253

(a) Not used in least squares fit

Least squares fits through the above scaling factors resulted in the following relationships:

Component	Least-Squares Fit
Reactor vessel	0.211 + 2.260E-4 (MWt)
Vessel internals	0.283 + 1.772E-4 (MWt)
Reactor building	0.267 + 2.035E-4 (MWt)

The component "reactor building" is furthermore broken down into smaller components including the sacrificial shield (activated concrete and rebar), the pressure suppression chamber, piping systems, HVAC and electrical equipment, reactor refueling pools, contaminated concrete surfaces, and miscellaneous equipment. The HVAC system in the reactor building consists of ducting, blowers, heaters, coolers, and filters. The reactor refueling pools include the spent fuel storage pool, the reactor well, and the dryer and separator storage pool. Contaminated concrete surfaces excludes surfaces in the reactor refueling pools and the pressure suppression chamber. Calculated scaling factors for these assorted subcomponents are presented in Table A-30.

Cost factors are calculated by PNL as the ratio of the component cost to the total of the seven component costs. A least squares fit through the scaling factors for the sacrificial shield yields the relationship: SF = 0.418 + 1.726E-5(MWt), while that for contaminated concrete surfaces yields: SF = 0.837 + 2.334E-5(MWt). The least squares fit through miscellaneous equipment yields: SF = 0.231 + 2.284E-4(MWt). The least squares fit through all the plant-specific scaling factors, which are developed based upon cost-weighted averages of the reactor vessel, vessel internals, and reactor building scaling factors, yields: SF = 0.267 + 2.035E-4(MWt).

Other approaches may be reviewed by reference to the literature. DOE has in the past used an approach whereby decommissioning waste volumes are assumed to be linearly proportional to a plant's electrical capacity in MW(e) (Refs. 66, 67). Another study has noted that while waste volumes would increase with reactor power level, the increase would probably not be linear (Ref. 68).

In this study (Ref. 68), it was postulated that waste volumes could be extrapolated from waste volumes generated from decommissioning the Elk River reactor plant. The following relationship was postulated:

$$V_2/V_1 = (R_2/R_1)^{0.8} \qquad \text{(A-2)}$$

where:

V_2 = waste volume generated from decommissioning reactor plant of interest.
V_1 = waste volume generated from decommissioning Elk River plant.
R_2 = thermal power level of reactor plant of interest.
R_1 = thermal power level of Elk River plant.

This postulated relationship was extrapolated from a rule-of-thumb relationship used to estimate the cost of constructing a second reactor plant when the cost and capacity of a first reactor plant is known. The Elk River plant represents one of the few plant decommissioning activities that was decommissioned by dismantlement (as opposed to mothballing or some other technique). The Elk River plant was a small, natural circulation BWR (58.3 MW(t)), and operated for only 3.5 years. The waste generated from the Elk River decommissioning was divided into six groups as follows:

Group	Waste Volume (m^3)
Reactor pressure vessel	4.6
Internals	1.1
Externals	5.3
Biological shield	5.9
Rad. & contaminated matls.	1300.
Contaminated concrete	1300.
	2616.9

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Table A-30. Component-Specific Scaling Factors for BWR Reactor Building Materials

Power Plant	Plant Power Rating (MWt)	Pressure Sacrificial Shield	Suppression Chamber	Piping Systems	HVAC and Electrical	Reactor Refueling Pools	Contaminated Concrete Surfaces	Miscellaneous Equipment	Component-Specific Scaling Factor[a]
Vermont Yankee	1,593	0.693	0.531	0.480	0.934	0.964	0.941	0.586	0.611
Oyster Creek	1,600	0.725	0.607	0.482	0.785	0.938	0.865	0.632	0.620
Monticello	1,670	0.691	0.568	0.503	0.841	0.846	0.876	0.612	0.612
Cooper	2,381	0.806	0.638	0.720	0.799	0.974	0.802	0.724	0.748
Dresden 2 or 3	2,527	0.824	0.817	0.760	0.875	1.151	0.880	0.808	0.826
Peach Bottom 2 or 3	3,293	0.990	0.990	0.990	0.876	1.167	0.878	0.995	0.995
WNP-2	3,320	1.000	1.000	1.000	1.000	1.000	1.000	1.000	1.000
Cost Factor for Reference-Plant Building Component		0.1454	0.1588	0.4250	0.0635	0.0826	0.0282	0.0965	

[a] Calculated as the sum of the products of the individual scaling factors and the corresponding cost factors.

The activity of decommissioning waste from various sized plants was also extrapolated from Elk River data using the following relationship:

$$A_2 = (A'/f)(R_2/R')^{0.8} \qquad \text{(A-3)}$$

where R_2 has been defined above and:

A_2 = activity of decommissioning waste from power plant of interest;
A' = activity of decommissioning waste from Elk River; and
R' = thermal power level of Elk River plant (58.3 MW(t)).

The factor f is used as a means of expressing the function of activation equilibrium achieved in a reactor operating less than 40 years--i.e.:

$$f = (1-e^{-\lambda t})/(1-e^{-40\lambda}). \qquad \text{(A-4)}$$

The factor f was furthermore developed from the assumption that neutron flux levels do not vary significantly with reactor capacity, and from the mass balance equation:

$$\frac{dN}{dt} = \Sigma_a \, \phi - \lambda N \qquad \text{(A-5)}$$

where:

Σ_a = macroscopic absorption cross section;
λ = decay constant;
ϕ = neutron flux;
t = time of reactor operation; and
N = atom density.

Apparently setting t = 3.5 years and assuming that different sized reactors each operated for 40 years, the authors then projected activity and volume levels for different sized plants (Ref. 68). These projections are as follows (no differentiation is made for PWRs and BWRs):

Capacity (MW(t))	Volume (m^3)	Activity (Ci)*
Elk River (58.3)	2,617	10,245
1000	25,164	165,296
2000	44,287	295,170
3000	61,405	411,230
3600	70,459	471,270

*Does not include activity of miscellaneous contaminated materials.

These projections may be compared with the PNL projections. For the reference BWR (1155 MW(e), 3320 MW(t)), the projected waste volume and activity are 21,396 m^3 and 6,605,090 Ci. For the reference PWR (1175 MW(e), 3500 MW(t)), the projected waste volume and activity are 18,856 m^3 and 4,891,230 Ci.

Another study on power plant decommissioning was performed by AIF in "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives"

(Ref. 60). For a 1160 MW(e) (3,411 MW(t)) PWR, decommissioning was postulated to require 4 years and result in 9940 m^3 of waste disposed in a licensed disposal site plus 18,350 m^3 of waste disposed in a sanitary landfill. For a 1160 MW(e) BWR (3,579 MW(t)), decommissioning was again postulated to require 4 years and result in 23,703 m^3 of waste disposed in a licensed disposal site plus 5,352 m^3 of waste disposed in a sanitary landfill.

The AIF study also considered some variations in the cost projections that would result from smaller reactor plants. The study noted that the nuclear steam supply systems and balance-of-plant systems of a 550 MW(e) BWR are approximately 53% smaller than a 1160 MW(e) plant, while the reactor vessel is about 23% smaller. Thus, the disposal costs, which are a function of both waste volume and curie content, are expected to be smaller for the smaller plant (i.e., $319,000 vs. $408,000). For a 550 MW(e) PWR, the NSSS systems and balance-of-plant systems are approximately 53% smaller than a 1160 MW(e) plant, while the reactor vessel for a 550 MW(e) PWR is about 24% smaller. The total disposal cost, however, is projected to be greater for the 550 MW(e) plant ($609,000 vs. $591,000) due to higher neutron fluxes in the smaller vessel (Ref. 60).

A.7.2.2 Waste Streams and Activity Distributions

The reference decommissioning waste streams are generated for this study from Tables A-26 through A-30 in a mostly straightforward manner. These waste streams are listed in Table A-31. As shown, different waste streams are postulated for each type of plant--i.e., either the reference 1155 MW(e) BWR or the reference 1175 MW(e) PWR. Annual volumes are given assuming that decommissioning requires 4 years. Volumes are presented as-generated and prior to consideration of any additional processing measures (e.g., compaction for compressible trash, solidification for evaporator bottoms).

In addition, activated metals have been divided into a number of individual waste streams. The reactor core shrouds are identified as separate decommissioning waste streams. The remaining activated metal streams are divided into miscellaneous reactor internals and pressure vessel material. This approach is taken since the core shroud and other reactor internals are primarily composed of stainless steel alloys while the reactor vessel is composed of carbon steel. A different radionuclide distribution would be expected for the two types of metal. Gross radionuclide concentrations are also given as distributions as presented in Tables A-32 and A-33. The concentrations of the other decommissioning waste streams are presented as averages across the total waste stream.

The format of Tables A-32 and A-33 is similar to that of Table A-12. A series of concentration ranges is presented, and in each range the volume percentage of the waste having concentrations in the range is listed. Also listed is the average concentrations of the wastes falling into each range.

A.7.2.3 Proportionment to Different Sized Plants

The approach that will be taken in this report is to proportion the individual waste stream volumes generated from decommissioning a given reactor to the waste stream volumes calculated for the two reference PNL facilities. This approach is at best an approximation, since it relies on at least three somewhat crude assumptions:

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Table A-31. Reference Waste Streams Assumed from Nuclear Power Plant Decommissioning

Waste Stream	Description	Annual Volume (m^3/yr)	Concentration (Ci/m^3)
Reference 1155 MW(e) BWR:			
B-DECORES	Activated core shroud	11.75	1.3E+5
B-DEACINT	Activated reactor internals	20.75	*
B-DEACVES	Activated reactor vessel	2.0	2.7E+2
B-DEACTCO	Activated concrete	22.5	1.9E+0
B-DECONME	Contaminated metal	3657	5.7E-1
B-DECONCO	Contaminated concrete	650	4.4E-2
B-DETRASH	Combustible/compactible trash	848	5.3E-1
B-DERESIN	Chelated ion exchange resins	10.5	5.4E+0
B-DEEVAPB	Evaporator bottoms	109.5	7.5E+1
		5332.**	
Reference 1175 MW(e) PWR:			
P-DECORES	Activated core shroud	2.75	3.1E+5
P-DEACINT	Activated reactor internals	63.	*
P-DEACVES	Activated reactor vessel	55.5	*
P-DEACTCO	Activated concrete	177.	2.8E+0
P-DECONME	Contaminated metal	1366.	1.6E-1
P-DECONCO	Contaminated concrete	2653.	9.4E-3
P-DETRASH	Combustible/compactible trash	354.5	5.3E-1
P-DERESIN	Chelated ion exchange resins	7.5	1.4E+3
P-DEFILCR	Filter cartridges	2.2	5.6E+2
P-DEEVAPB	Evaporator bottoms	33.	1.0E+2
		4714.**	

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*Given as concentration distributions in Tables A-32 and A-33.

**Total projected as-generated waste volumes over 4 years are 21,328 m^3 for the reference BWR and 18,856 m^3 for the reference PWR.

Table A-32. Concentration Distribution Across Activated Reactor Internals Waste Streams (DEACINT)

	Reference BWR		Reference PWR	
Range (Ci/m^3)	Volume Percent	Average Conc. (Ci/m^3)	Volume Percent	Average Conc. (Ci/m^3)
.5 $\leq$ 1			9.9	0.8
1 $\leq$ 5				
5 $\leq$ 10			11.1	7.4
10 $\leq$ 50	4.8	25.		
50 $\leq$ 100	13.3	59.	21.4	84.
100 $\leq$ 500	6.0	140.	3.6	236.
500 $\leq$ 1,000	12.0	960.		
1,000 $\leq$ 5,000	45.8	1,340.	5.6	1,740.
5,000 $\leq$ 10,000			42.9	7,840.
> 10,000	18.1	12,600.	5.6	39,500.
Weighted Average:		3,030.		5,700.

Table A-33. Concentration Distribution Across Activated PWR Reactor Vessel Waste Stream (P-DEACVES)

Range (Ci/m^3)	Volume Percent	Average Conc. in Range (Ci/m^3)
.1 $\leq$ .5	51.4	0.18
.5 $\leq$ 1		
1 $\leq$ 5		
5 $\leq$ 10		
10 $\leq$ 50		
50 $\leq$ 100		
100 $\leq$ 500	48.6	178
Weighted Average:		86.6

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1. Decommissioning activities at any size plant will involve approximately the same types of waste streams.
2. All plants are designed approximately the same.
3. All plants have somewhat similar operating histories.

The first assumption is believed to be the most reasonable of the three. The second assumption is believed to be more reasonable for newer plants than for older plants, which often featured somewhat novel designs. The newer plants have been much more standardized in an effort to reduce costs. The third assumption is also believed to be more reasonable for newer plants than for older ones. This is because many of the newer plants were experimental, low power units, and also most of the best candidates for decommissioning in a fairly recent time have not operated for very many years. The number of full power years that a plant operates will have an influence on the volume of activated and contaminated material that is produced.

Notwithstanding the uncertainties, the approach taken is to first assume that the waste streams characterized in Table A-31 are common to all power plant decommissioning projects. Each decommissioning project is assumed to require 4 years to complete, and the volumes of each waste stream are scaled from the volumes projected from the two reference reactor plants. The scaling factors are obtained from the PNL data.

It is believed that using the PNL scaling factors may involve taking the calculated relationships further than the authors of the PNL studies originally intended. However, the authors of this report can find no better data upon which to project waste stream volumes. In any case, it is believed that scaling each waste stream is the preferred approach, and once the overall format is laid out, perhaps improved general relationships may be developed in time.

The scaling factors are of the following forms:

$$SF_p = A + B\,(MWt) + C\,(MWt)^2 + D\,(MWt)^3 \qquad \text{(A-6)}$$

$$SF_B = E + F\,(MWt) \qquad \text{(A-7)}$$

where:

SF_p = scaling factors for PWR decommissioning waste streams;
SF_B = scaling factors for BWR decommissioning waste streams; and
MWt = reactor thermal capacity in MW.

The constants A through D are listed in Table A-34 for each PWR waste stream while the constants E and F are listed in Table A-35 for each BWR waste stream.

For PWRs, the scaling factors for the P-DECORES and P-DEACINT waste streams were taken from the PNL "Vessel Internals" scaling factors (see background section), while the P-DEACVES waste stream uses the PNL "Reactor Vessel" scaling factors. The P-DEACTCO and P-DECONCO waste streams use the PNL "Containment Building" scaling factors. The P-DECONME waste stream uses a set of scaling factors based on a cost-weighted average of the PNL scaling factors for "pumps

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Table A-34. Scaling Factor Constants for PWR Decommissioning Waste Streams

Waste Stream	Constants			
	A	B	C	D
P-DECORES	4.000 E-1	2.318 E-4	-2.397 E-8	1.925 E-12
P-DEACINT	4.000 E-1	2.318 E-4	-2.397 E-8	1.925 E-12
P-DEACVES	1.300 E-1	3.589 E-4	-2.713 E-8	-1.258 E-12
P-DEACTCO	2.300 E-1	5.829 E-5	4.163 E-8	1.307 E-12
P-DECONME	5.080 E-1	-3.917 E-4	3.156 E-7	-4.673 E-11
P-DECONCO	2.300 E-1	5.829 E-5	4.163 E-8	1.307 E-12
P-DETRASH	-1.145 E-2	5.954 E-4	-1.757 E-7	2.518 E-11
P-DERESIN	-1.145 E-2	5.954 E-4	-1.757 E-7	2.518 E-11
P-DEFILCR	-1.145 E-2	5.954 E-4	-1.757 E-7	2.518 E-11
P-DEEVAPB	-1.145 E-2	5.954 E-4	-1.757 E-7	2.518 E-11

Table A-35. Scaling Factor Constants for BWR Decommissioning Waste Streams

Waste Streams	Constants	
	E	F
B-DECORES	2.830 E-1	1.772 E-4
B-DEACINT	2.830 E-1	1.772 E-4
B-DEACVES	2.110 E-1	2.260 E-4
B-DEACTCO	4.180 E-1	1.726 E-5
B-DECONME	0.	3.106 E-4
B-DECONCO	8.370 E-1	2.334 E-5
B-DETRASH	2.670 E-1	2.035 E-4
B-DERESIN	2.670 E-1	2.035 E-4
B-DEEVAPB	2.670 E-1	2.035 E-4

and piping," "stream generators," and "pressurizers." All the remaining PWR waste streams use a set of cost-weighted scaling factors derived from all of the listed components.

Similarly, BWR waste streams involving reactor internals or the reactor pressure vessel are scaled using the PNL scaling factors for similar components. The contaminated and activated concrete waste streams are likewise based on PNL scaling factors for similar materials, while the B-DECONME waste stream is based on PNL scaling factors for BWR "piping systems" (see Table A-29). The remaining waste streams are scaled across all plant components.

Use of the above approach can be illustrated by the Elk River example, a 58.3 MW(t) BWR. Projected volumes are as follows, assuming that trash is compacted by a factor of 2, resins are dewatered, and evaporator bottoms are solidified in cement:

Waste Stream	Ref. Ann. Vol. (m^3/yr)	Decom. yrs.	VRF	VIF	Scaling Factor	Projected Elk River Vol. (m^3)
B-DECORES	11.75	4	1.0	1.0	.293	13.8
B-DEACINT	20.75	4	1.0	1.0	.293	24.3
B-DEACVES	2.	4	1.0	1.0	.224	1.8
B-DEACTCO	22.5	4	1.0	1.0	.419	37.7
B-DECONME	3657.	4	1.0	1.0	.018	263.3
B-DECONCO	650.	4	1.0	1.0	.838	2178.8
B-DETRASH	848.	4	2.0	1.0	.279	473.2
B-DERESIN	10.5	4	1.0	1.0	.279	11.7
B-DEEVAPB	109.5	4	1.0	1.4	.279	171.1
						3175.7

From the background information presented earlier, it can be seen that the projected volume is about 21% higher than the actual decommissioning value (2620 m^3). A more optimistic volume reduction factor of 6 for trash would reduce the margin between projected and actual volumes to about +9%. A strictly linear approach would have resulted in a projected 376 m^3, a factor of 7 below the actual volume.

A.7.2.4 Timing of Power Plant Decommissioning

Background

Forty calendar years of operating life is generally considered an appropriate assumption for the length of service life of a large modern LWR prior to decommissioning. Based upon this assumption, Table A-36 was generated, illustrating a number of reactors which can be postulated to be candidates for decommissioning in the general neighborhood of the year 2000. Using this criteria, only two reactors -- Shippingport and Dresden 1 -- would be projected for decommissioning prior to the year 2000. However, as discussed below, such projections are uncertain and may depend upon factors other than the assumed 40-year operating life of the units.

Table A-36. Power Reactors Theoretically Eligible for Decommissioning After Forty Years Operation

Name	State Located	Type	Power MW(e)	Power MW(t)	Start-up	Theoretical Shutdown
Shippingport	PA (I)*	PWR	60	236	1957	1997
Dresden 1	IL (III)	BWR	200	700	1959	1999
Yankee Rowe	MA (I)	PWR	175	600	1960	2000
Indian Point 1	NY (I)	PWR	265	615	1962	2002
Big Rock Point	MI (III)	BWR	72	240	1962	2002
Humboldt Bay	CA (V)	BWR	65	242	1963	2003
Haddam Neck	CN (I)	PWR	575	1825	1967	2007
LaCrosse	WI (III)	BWR	50	165	1967	2007
San Onofre	CA (V)	PWR	436	1347	1967	2007
Oyster Creek	NJ (I)	BWR	650	1930	1969	2009
Nine Mile Point 1	NY (I)	BWR	620	1850	1969	2009
R. E. Ginna 1	NY (I)	PWR	470	1520	1969	2009
Millstone 1	CN (I)	BWR	660	2011	1970	2010
H. B. Robinson	SC (II)	PWR	700	2200	1970	2010
Dresden 2	IL (III)	BWR	794	2527	1970	2010
Monticello	MN (III)	BWR	545	1670	1970	2010
Point Beach 1	WI (III)	PWR	497	1518	1970	2010
Dresden 3	IL (III)	BWR	794	2527	1971	2011
Palisades	MI (III)	PWR	805	2530	1971	2011
Maine Yankee	ME (I)	PWR	825	2630	1972	2012
Vermont Yankee	VT (I)	BWR	514	1593	1972	2012
Surry 1	VA (II)	PWR	822	2441	1972	2012
Turkey Point 3	FL (II)	PWR	693	2200	1972	2012
Point Beach 2	WI (III)	PWR	497	1518	1972	2012
Quad-Cities 1	IL (III)	BWR	789	2511	1972	2012
Quad-Cities 2	IL (III)	BWR	789	2511	1972	2012
Indian Point 2	NY (I)	PWR	873	2758	1973	2013
Peach Bottom 2	PA (I)	BWR	1065	3293	1973	2013
Browns Ferry 1	AL (II)	BWR	1065	3293	1973	2013
Oconee 1	SC (II)	PWR	887	2568	1973	2013
Oconee 2	SC (II)	PWR	887	2568	1973	2013
Surry 2	VA (II)	PWR	822	2441	1973	2013
Turkey Point 4	FL (II)	PWR	693	2200	1973	2013
Prairie Island 1	MN (III)	PWR	530	1650	1973	2013
Zion 1	IL (III)	PWR	1040	3250	1973	2013
Zion 2	IL (III)	PWR	1040	3250	1973	2013
Ft. Calhoun	NE (IV)	PWR	457	1500	1973	2013
Calvert Cliffs 1	MD (I)	PWR	845	2700	1974	2014
Calvert Cliffs 2	MD (I)	PWR	845	2700	1974	2014
J. A. Fitzpatrick	NY (I)	BWR	821	2436	1974	2014
Peach Bottom 3	PA (I)	BWR	1065	3293	1974	2014
Pilgrim 1	MA (I)	BWR	665	1998	1974	2014
Three Mile Island 1	PA (I)	PWR	819	2535	1974	2014
Browns Ferry 2	AL (II)	BWR	1065	3293	1974	2014

Table A-36. Power Reactors Theoretically Eligible for Decommissioning After Forty Years Operation (Continued)

Name	State Located	Type	Power		Start-up	Theoretical Shutdown
			MW(e)	MW(t)		
E. I. Hatch 1	GA (II)	BWR	786	2436	1974	2014
Oconee 3	SC (II)	PWR	887	2568	1974	2014
Duane Arnold 1	IA (III)	BWR	538	1658	1974	2014
Kewaunee	WI (III)	PWR	535	1650	1974	2014
Prairie Island 2	MN (III)	PWR	530	1650	1974	2014
Arkansas 1	AR (IV)	PWR	850	2568	1974	2014
Cooper	NE (IV)	BWR	778	2381	1974	2014
Ft. St. Vrain	CO (IV)	HTGR	330	842	1974	2014
Rancho Seco 1	CA (V)	PWR	918	2772	1974	2014
Millstone 2	CT (I)	PWR	870	2700	1975	2015
Brunswick 2	NC (II)	BWR	821	2436	1975	2015
D. C. Cook 1	MI (III)	PWR	1054	3250	1975	2015
Trojan 1	OR (V)	PWR	1130	3411	1975	2015
Beaver Valley 1	PA (I)	PWR	852	2660	1976	2016
Indian Point 3	NY (I)	PWR	965	3025	1976	2016
Salem 1	NJ (I)	PWR	1090	3338	1976	2016
Browns Ferry 3	AL (II)	BWR	1065	3293	1976	2016
Brunswick 1	NC (II)	BWR	821	2436	1976	2016
St. Lucie 1	FL (II)	PWR	802	2700	1976	2016
Crystal River 3	FL (II)	PWR	825	2544	1977	2017
J. M. Farley 1	AL (II)	PWR	829	2652	1977	2017
Davis-Besse 1	OH (III)	PWR	906	2772	1977	2017
E. I. Hatch 2	GA (II)	BWR	784	2436	1978	2018
North Anna 1	VA (II)	PWR	907	2775	1978	2018
D. C. Cook 2	MI (III)	PWR	1100	3391	1978	2018
Three Mile Island 2	PA (I)	PWR	906	2772	1979	2019
North Anna 2	VA (II)	PWR	907	2775	1980	2020
Arkansas 2	AR (IV)	PWR	912	2815	1980	2020
Salem 2	NJ (I)	PWR	1115	3411	1981	2021
J. M. Farley 2	AL (II)	PWR	829	2652	1981	2021
Sequoyah 1	TN (II)	PWR	1148	3411	1981	2021
W. B. McGuire 1	NC (II)	PWR	1180	3411	1981	2021
Sequoyah 2	NC (II)	PWR	1148	3411	1982	2022
La Salle 1	IL (III)	BWR	1078	3293	1982	2022
San Onofre 2	CA (V)	PWR	1100	3410	1983	2023
Susquehanna 1	PA (I)	BWR	1050	3293	1983	2023
St. Lucie 2	FL (II)	PWR	810	2570	1983	2023
San Onofre 3	CA (V)	PWR	1100	3410	1984	2024
WNP-2	WA (V)	BWR	1100	3323	1984	2024
La Salle 2	IL (III)	BWR	1078	3293	1984	2024
V. C. Summer 1	SC (II)	PWR	900	2785	1984	2024
W. B. McGuire 2	NC (II)	PWR	1180	3411	1984	2024
Diablo Canyon 1	CA (V)	PWR	1084	3338	1985	2025
Watts Bar 1	TN (II)	PWR	1177	3425	1985	2025
Shoreham	NY (I)	BWR	819	2436	1985	2025

Table A-36. Power Reactors Theoretically Eligible for Decommissioning After Forty Years Operation (Continued)

Name	State Located	Type	Power MW(e)	Power MW(t)	Start-up	Theoretical Shutdown
Susquehanna 2	PA (I)	BWR	1050	3293	1985	2025
Grand Gulf 1	MS (II)	BWR	1250	3833	1985	2025
Waterford 3	LA (II)	PWR	1113	3410	1985	2025
Byron 1	IL (III)	PWR	1120	3425	1985	2025
Callaway 1	MO (III)	PWR	1120	3411	1985	2025
E. Fermi 2	MI (III)	BWR	1093	3292	1985	2025
Comanche Peak 1	TX (IV)	PWR	1111	3411	1985	2025
Palo Verde 1	AZ (V)	PWR	1270	3817	1985	2025
Limerick 1	PA (I)	BWR	1065	3293	1985	2025
Catawba 1	SC (II)	PWR	1145	3411	1985	2025
River Bend 1	LA (II)	BWR	934	2894	1985	2025
Perry 1	OH (III)	BWR	1205	3579	1985	2025
Wolf Creek	KS (IV)	PWR	1150	3411	1985	2025
Diablo Canyon 2	CA (V)	PWR	1106	3411	1985	2025
Hope Creek 1	NJ (I)	BWR	1067	3293	1986	2026
Clinton 1	IL (III)	BWR	933	2894	1986	2026
Nine Mile Point 2	NY (I)	BWR	1100	3323	1986	2026
Beaver Valley 2	PA (I)	PWR	833	2660	1986	2026
Shearon Harris 1	NC (II)	PWR	900	2775	1986	2026
Braidwood 1	IL (III)	PWR	1120	3360	1986	2026
Comanche Peak 2	TX (IV)	PWR	1111	3411	1986	2026
Palo Verde 2	AZ (V)	PWR	1270	3817	1986	2026
Millstone 3	CT (I)	PWR	1156	3411	1986	2026
Byron 2	IL (III)	PWR	1120	3425	1986	2026
Seabrook 1	NH (I)	PWR	1200	3411	1986	2026
A. W. Vogtle 1	GA (II)	PWR	1110	3425	1987	2027
Catawba 2	SC (II)	PWR	1145	3411	1987	2027
Braidwood 2	IL (III)	PWR	1120	3425	1987	2027
South Texas 1	TX (IV)	PWR	1250	3817	1987	2027
Palo Verde 3	AZ (V)	PWR	1270	3817	1987	1027
A. W. Vogtle 2	GA (II)	PWR	1110	3425	1988	2028
Limerick 2	PA (I)	BWR	1065	3293	1988	2028
Watts Bar 2	TN (II)	PWR	1177	3425	1988	2028
Grand Gulf 2	MS (II)	BWR	1250	3833	1989	2029
Bellefonte 1	AL (II)	PWR	1213	3621	1989	2029
South Texas 2	TX (IV)	PWR	1250	3817	1989	2029
WNP 1	WA (V)	PWR	1218	3619	1989	2029
WNP 3	WA (V)	PWR	1242	3817	1989	2029
Perry 2	OH (III)	BWR	1205	3579	1990	2030
Bellefonte 2	AL (II)	PWR	1213	3621	1991	2031
Seabrook 2	NH (I)	PWR	1200	3411	1992	2032
Carroll Cty 1	IL (III)	PWR	1120	3425	2001	2041
Carroll Cty 2	IL (III)	PWR	1120	3425	2002	2042

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*NRC Region

The first seven plants listed (plus the La Crosse unit) are low power units generally constructed as demonstration projects forerunning units which are considerably larger and more economical to operate. Although utilities would generally prefer to keep the older units operable for as long as they are cost-effective, costs of upgrading the older units to meet new NRC safety requirements may result in some of the older plants being decommissioned prior to the year 2000, and prior to the end of their otherwise serviceable lives. Short discussions of the present and possible future status of these early units follow:

- Shippingport. The first nuclear power reactor constructed, this unit has been recently operated by the Department of Navy as a research reactor. This reactor is currently shut down and is planned for decommissioning as a demonstration project. It is anticipated that the waste that will be generated from this project will be disposed at DOE rather than commercial facilities.

- Dresden 1. This unit was the first BWR built for commercial use. This plant has been shut down for several years, and full scale decontamination of the primary cooling system has been carried out. The plant operators, Commonwealth Edison, had at one time indicated that they would not make a decision on restarting the plant until June 1986. Dresden 1 is not a particularly large unit, but a number of modifications would be required to restart the unit based upon new NRC requirements imposed since the TMI-2 accident. Commonwealth Edison was uncertain whether it would be economically worthwhile to make the modifications and restart the unit (Ref. 69). More recently (Ref. 70), Commonwealth Edison has decided to retire the unit. It will be mothballed for the remaining useful life of the entire site, which has two other operating units. All three units are therefore assumed to be dismantled at about the same time.

- Yankee-Rowe. This unit continues to generate electricity with no apparent problems. In 1979, its capacity factor was 81% (Ref. 1).

- Indian Point 1. This unit (IP1) was shut down in October 1974 by its utility, Consolidated Edison, due to inability to meet new NRC requirements on emergency core cooling systems (ECCS). Consolidated Edison has recently determined that the cost of upgrading the plant to meet the new ECCS and other requirements would be greatly in excess of the possible economic gain, and has announced their intention of decommissioning the unit. Commonwealth Edison has proposed to mothball IP1 until 2006, when the adjacent Indian Point, Unit 2 (IP2) is tentatively scheduled for decommissioning. Both units would then be decommissioned at the same time. The units share a number of activities, and Commonwealth Edison has taken a position that the operation of IP2 is dependent upon the physical existence of IP1. Under the current IP1 license Commonwealth Edison is authorized to store spent fuel, perform radioactive waste processing for IP2, and other related activities (Ref. 71).

- Big Rock Point. This BWR is presently in operation. The utility, Consumers Power Company, would prefer to keep the unit in operation as long as it is economical to do so (Ref. 72).

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- Humboldt Bay. This unit has been shut down since July 1976 for refueling, maintenance, seismic modifications, and area geologic studies requested by NRC. The operators are Pacific Gas and Electric (PG&E), who have proposed to decommission the unit by putting it into safe storage for 30 years, after which it will be dismantled. The reason given by PG&E that immediate dismantlement is not preferred is that there is no place to store the spent fuel except onsite. PG&E essentially proposes to keep the plant in safe storage until DOE is in a position to accept fuel either for storage in an away-from-reactor (AFR) facility, or disposal in a geologic repository (Ref. 73).
- La Crosse (Genoa). This unit is currently operating. Earlier questions regarding the potential for liquefaction of site soil during a design basis earthquake have been resolved. The utility, Dairyland Power Cooperative, would like to continue running the plant while it is economical to do so (Ref. 74).

Projections

Despite the many uncertainties, a projection of light water reactor decommissioning schedules is needed in order to estimate the impacts and costs associated with disposal of decommissioning wastes. In this report, decommissioning waste volumes are assumed to be proportional to a reactor's thermal power capacity (in MWt) as discussed in the previous section. The assumed LWR decommissioning schedules are presented in terms of thermal power capacity in Table A-37 for PWR plants and in Table A-38 for BWR plants.

The schedules are tabulated principally based upon an assumption that reactor dismantlement takes place 40 years following startup of the unit. However, little or no dismantlement is assumed to take place prior to the year 2000. This is the approximate time that a Federal repository or storage facility is projected to be available and accepting spent light water reactor fuel. Few utilities are believed to be willing to dismantle reactor units until there is a definite process in place for Federal acceptance of spent reactor fuel. One reactor currently shut down (Humboldt Bay) is assumed to be held in safe storage until the year 2000. The Dresden 1 unit is also currently shut down and will be mothballed until it is decommissioned and dismantled at the same time as Units 2 and 3 (see Section A.4.2). Indian Point 1 is assumed to be held in storage until Unit 2 is decommissioned. The remaining units are assumed to be decommissioned in the order given in Table A-36.

A.7.3 West Valley Demonstration Project

Wastes generated as part of the West Valley Demonstration Project are not regulated directly by NRC (i.e., DOE is not an NRC licensee). However, much of the waste that will be generated will be suitable for near-surface disposal, and DOE has an environmental assessment under preparation in which low-level waste disposal alternatives are being considered. Some of the disposal alternatives being considered include (1) disposal at a regional disposal facility sited by a compact, and (2) disposal into the existing NRC-licensed disposal area located on the West Valley facility. Thus, the potential impacts of disposal of the waste are of interest vis-a-vis the requirements in 10 CFR Part 61.

Table A-37. Projected PWR Decommissioning Schedule (MW(t))

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
2000	Yankee Rowe	600								
2001										
2002										
2003										
2004										
2005										
2006										
2007	Haddam Neck	1825							San Onofre 1	1347
2008										
2009	R.E. Ginna	1520								
2010			H.B. Robinson	2200	Point Beach 1	1518				
2011					Palisades	2530				
2012	Maine Yankee	2630	Surry 1 Turkey Pt 3	2441 2200 4,641	Pt Beach 2	1518				
2013	Indian Pt 1 Indian Pt 2	615 2758 3,373	Oconee 1 Oconee 2 Surry 2 Turkey Pt 4	2568 2568 2441 2200 9,777	Prairie Is. 1 Zion 1 Zion 2	1650 3250 3250 8,150	Ft Calhoun	1500		

Table A-37. (continued)

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
2014	Cal. Cliffs 1	2700	Oconee 3	2568	Kewaunee	1650	Arkansas 1	2568	Rancho Seco	2772
	Cal. Cliffs 2	2700			Prairie Is. 2	1650				
	TMI-1	2535				3,300				
		7,935								
2015	Millstone 2	2700			DC Cook 1	3250			Trojan 1	3411
2016	Beaver Val. 1	2660	St. Lucie 1	2700						
	Indian Pt 3	3025								
	Salem 1	3338								
		9,023								
2017			Crystal River 3	2544	Davis-Besse	2772				
			J.M. Farley 1	2652						
				5,196						
2018			North Anna 1	2775	D.C. Cook 2	3391				
2019	TMI-2	2772								
2020			North Anna 2	2772	Arkansas 2	2815				
2021	Salem 2	3411	J.M. Farley 2	2652						
			Sequoyah 1	3411						
			W.B. McGuire 1	3411						
				9,474						
2022			Sequoyah 2	3411						
2023			St. Lucie 2	2570					San Onofre 2	2023
2024			V.C. Summer	2785						
			W.B. McGuire	3411					San Onofre 3	3410
				6,196						

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Table A-37. (continued)

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
2025			Catawba 1	3411	Byron 2	3425	Com. Pk 1	3411	Palo Verde 1	3817
			Waterford 3	3410	Callaway 1	3411	Wolf Creek	3411	Diablo Can. 1	3338
			Watts Bar 1	3425		6,836		6,822	Diablo Can. 2	3411
				10,246						10,566
2026	Beaver Val. 2	2660	S. Harris 1	2775	Braidwood 1	3360	Com. Pk 2	3411	Palo Verde 2	3817
	Millstone 3	3411			Byron 2	3425				
	Seabrook 1	3411				6,785				
		9,482								
2027			A.W. Vogtle 1	3425	Braidwood 2	3425	S. Texas 1	3817	Palo Verde 3	3817
			Catawba 2	3411						
				6,836						
2028			A.W. Vogtle 2	3425						
			Watts Bar 2	3425						
				6,850						
2029			Bellefonte 1	3621			S. Texas 2	3817	WNP 1	3619
									WNP 3	3817
										7,436
2030			Bellefonte 2	3621						
2031										
2032	Seabrook 2	3411								
2033										
2034										
2035										
2036										

Table A-37. (continued)

Year	Region 1	Region 2	Region 3	Region 4	Region 5
2037					
2038					
2039					
2040					
2041			Car. Cty 1 3425		
2042			Car. Cty 2 3425		

Table A-38. Projected BWR Decommissioning Schedule (MW(t))

Year	Region 1		Region 2		Region 3		Region 4	Region 5	
2000								Humboldt Bay	242
2001									
2002					Big Rock Pt	240			
2003									
2004									
2005									
2006									
2007					La Crosse	165			
2008									
2009	Oyster Creek Nine Mile Pt 1	1930 1850 3,780							
2010	Millstone 2	2011			Monticello	1670			
2011					Dresden 1 Dresden 2 Dresden 3	700 2527 2527 5,754			
2012	Vt. Yankee	1593			Quad-Cities 1 Quad-Cities 2	2511 2511 5,022			
2013	Peach Bott. 2	3293	Browns Ferry 1	3293					

Table A-38. (continued)

Year	Region 1		Region 2		Region 3		Region 4		Region 5	
2014	Fitzpatrick	2436	Browns Ferry 2	3293	Duane Arnold	1658	Cooper	2381		
	Peach Bott. 3	3293	E.I. Hatch 1	2436						
	Pilgrim	1998		5,729						
		7,727								
2015			Brunswick 2	2436						
2016			Browns Ferry 3	3293						
			Brunswick 1	2436						
				5,729						
2017										
2018			E.I. Hatch 2	2436						
2019										
2020										
2021										
2022					La Salle 1	3293				
2023	Susquehanna 1	3293								
2024					La Salle 2	3293			WNP-2	3323
2025	Limerick 1	3293	Grand Gulf 1	3833	E. Fermi 2	3292				
	Susquehanna 2	3293	River Bend 1	2894	Perry 1	3579				
	Shoreham	2436		6,727		6,871				
		9,022								
2026	Hope Creek 1	3293			Clinton 1	2894				
	Nine Mile Pt 2	3323								
		6,616								

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Table A-38. (continued)

Year	Region 1		Region 2		Region 3		Region 4	Region 5
2027								
2028	Limerick 2	3293						
2029			Grand Gulf 2	3833				
2030					Perry 2	3579		

A.7.3.1 Background

In this report, the waste characteristics and projections are developed based on the assumed implementation of the preferred Alternative 1a (separated salt/sludge, onsite processing to terminal waste form) as discussed in the West Valley FEIS prepared by DOE (Ref. 75). Briefly, two general types of high-level liquid waste are currently being stored: Thorex waste and Purex waste.

About 47,000 liters of liquid waste are stored in Tank 8D-3; this waste was generated during a single processing campaign (Thorex process) wherein uranium was recovered from an experimental reactor fuel originally composed of 93.5% thorium and 6.5% highly enriched uranium. This waste is highly acidic and is thought to be principally in a single-phase acidic solution consisting primarily of thorium nitrate. It is possible, however, that some of the liquid waste has precipitated over time and has formed a bottom sludge. More significantly, about 2 million liters of liquid waste are stored in Tank 8D2 in a neutralized chemical form. This waste was generated as part of the majority of the processing operations (Purex process) and consists of a layer of supernate over a thick (about 1.4 m) layer of sludge.

In the operation originally proposed in the DOE FEIS (Ref. 75), the acidic Thorex waste would be removed from the storage tank and processed into a solid form. It is expected that the existing reprocessing facility, with modifications, can be used for solidification. (Prior to making such modifications, parts of the existing facility will be decontaminated and the existing contaminated equipment removed.) Then the neutralized Purex waste would be homogenized (i.e., the sludge would be mixed with the overlying supernatant liquid), pumped in batches to a holding tank, and thence to the main process building. The sludge would again be separated from the liquid and the supernate passed through an ion-exchange column. The sludge and ion-exchange media would be mixed together and incorporated into a terminal high-level waste form, probably borosilicate glass. The treated filtrate would be evaporated to a salt cake and solidified in cement.

Using this approach, approximately 300 HLW containers would be generated, of which 82 would consist of Thorex HLW (waste load 15%) and the remainder Purex HLW (waste load 25%). Each canister would be about 2 ft in diameter by 10 ft high, and would contain about 28.3 ft^3 of solidified waste (9 ft fill height). This would result in a total solidified waste volume of about 8500 ft^3 (240.3 m^3) being generated over 3 years of operation.

The HLW canisters would be stored on site until a HLW repository became available. The empty high-level liquid waste tanks would be decontaminated with acids, filled with cement, and left onsite. (Alternatively, the tanks could be dismantled and disposed as waste.) The facilities used to solidify the high-level waste would also be decontaminated and decommissioned, probably by dismantlement.

It should be noted that all of the operational decisions connected with the project have not yet been made, and will not be until further along in the project. A number of variations in the above operations are being considered. For example, the current plan is apparently to blend the Thorex and Purex waste prior to solidification. This of course increases the difficulty of making waste projections.

In any event, operations have started with an initial program in which spent fuel currently being stored in the reprocessing plant is being returned to the generators. After this, various decontamination and partial dismantlement (removal of some contaminated equipment) operations will be carried out, followed by construction of HLW solidification facilities. After solidification, the solidification facilities will be decontaminated and decommissioned. All this will require several years of effort, and will result in several thousand cubic meters of low activity waste being generated. Some of this waste is projected to be in concentrations that exceed Part 61 Class C limits, primarily because of transuranic content. Typical waste forms projected to be generated include trash and other low activity wastes (both compactible and noncompactible), ion-exchange resins and filter media, liquids and evaporator bottoms, and contaminated scrap equipment and hardware. Decontamination and decommissioning activities will also produce many of the above types of waste plus scrap fuel storage racks and demolition rubble (principally contaminated concrete).

A.7.3.2 Waste Projections

Projections of waste are made primarily using two documents (Refs. 75, 76). The DOE FEIS (Ref. 75) provides a projected operational schedule (Table A-39) plus a table (Table B.16 in the FEIS) listing projected volumes of different types of non-high-level wastes. This table also estimates the fraction projected to contain transuranics in concentrations exceeding 10 nCi/gm. Reference 76 provides an update of these projections, including projected radionuclide activities, as well as some information as to the timing of waste generation. Reference 76 limits its consideration to waste having a transuranic content less than 100 nCi/gm.

Waste types and overall as-packaged volumes are listed in Table A-40. A number of individual waste types (trash equipment) may be generated by more than one activity, and several types of waste may also contain varying levels of activity (hence, the waste volumes grouped into waste classes A, B, or C). In addition, some of the waste types may also be generated in volumes that exceed 100 nCi/gm in transuranic content. Finally, some types of waste will be generated at different points in time. Existing trash and dry solids will be generated until the end of vitrification on June 30, 1990, at which time trash from post-solidification decontamination and decommissioning (D/D) will dominate. Waste from the existing low-level waste treatment facility (LLWTF) is projected to be generated until operation of the radwaste treatment system (RTS) beginning on June 20, 1987. Similarly, waste from the fuel receiving and storage facility (FRS) is projected to be generated until the last spent fuel is shipped out on September 30, 1985.

The result from Table A-40 is a total of 43 potential waste streams, not including solidified high-level waste and lower activity waste streams that may have concentrations exceeding Class C limits. These latter waste streams are, or are expected to be, principally contaminated with transuranic radionuclides. These wastes are being stored pending a decision on their disposition, which is expected to involve shipment to a DOE facility for eventual disposal.

For completeness, all waste streams should be included in the data base. This means that some work is required to condense the above wastes into a more

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Table A-39. Schedule Assumed for Purposes of Analysis of Alternative 1a*

Activity	Start	Finish
Preconceptual Design Studies	1982	1983
Construction/Modifications:		
Temporary low-level waste storage facility		1984
Temporary high-level waste storage facility		1987
Decontamination of existing facilities	1983	1985
Installation and checkout of equipment	1985	1987
Operations:		
Processing wastes to terminal form	1987	1990
Decontamination and Decommissioning:		
Shipment of spent fuel offsite		1992
Decontamination of tanks and entombment with cement		1992
Dismantlement of processing facility	1992	1996
Shipment of high-level and transuranic wastes to a Federal repository	1997	2000
Dismantlement of temporary high-level waste storage facility		2000
Dismantlement of temporary low-level waste storage facility		2000
Dismantlement of low-level waste treatment facility		2000

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*These dates were used for purposes of analysis in reference 75. However, they may change as a result of ongoing engineering design and project planning.

Table A-40. Summary of Projected Low-Level Waste Volumes from West Valley Demonstration Project (< 100 nCi/gm TRU)

Waste Type	Description	Class	As-Treated Volume (m^3)	Treatment
Existing Systems Operations:				
Trash	Cloth, paper, plastic, etc.	A	1,800	3:1 compaction
Misc. dry solids	Soil, wood, railroad ties, concrete, etc.	A	5,700	none
Wet solid	LLWTF(a) sludge: Belltile clay, polyelectrolyte, sulfates, and nitrates	A	360	cement solid.
Wet solid	LLWTF resin	A	14	cement solid.
Wet solid	FRS(b) filter precoat: diatomaceous earth and Zeolon-100	B	6	cement solid.
Wet solid	FRS resin	B	39	cement solid.
RTS(c) Secondary Waste Streams:				
Liquid	Eluate and regenerate from polishers, 0.4 w/o $NaNO_3$ and 1.13 w/o HNO_3	B	25	cement solid.
Wet solid	Spent resin slurry	B	6	cement solid.
Wet solid	5 w/o filter backash slurry	B	25	cement solid.
Presolidification Decontamination Wastes:				
Trash	Cloth, paper, plastic, etc.	A	51	3:1 compaction
		B	15	3:1 compaction
		C(d)	33	3:1 compaction
Equipment & hardware	Predominantly steel and stainless steel	B	1,100	none
		C	110	none
		C(d)	80	none
Liquid	6.5 w/o salt before treatment	C	92	cement solid.
Secondary Waste Streams From Vitrification:				
Trash	Cloth, paper, plastic, etc.	A	140	3:1 compaction
		B	42	3:1 compaction
		C(d)	93	3:1 compaction
Failed Equipment	Predominantly stainless steel and Inconel	B	38	none
		C	4	none
		C(d)	3	none
Liquid	Decontaminated supernatent	A	2,100	cement solid.

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Table A-40. (continued)

Waste Type	Description	Class	As-Treated Volume (m^3)	Treatment
Liquid	Sludge wash solution, similar to supernatent, but 2 w/o salt	C	130	cement solid.
Liquid	Submerged bed scrubber condensate 3M HNO_3 content equivalent to 20 w/o	C	790	cement solid.
Liquid	Melter feed concentrator overheads	C	11	cement solid.
Liquid	Fractionator condensate. 0.27 M HNO_3 or 1.66 w/o HNO_3 with $NaNO_3$ content equivalent to 2.24 w/o.	A	2	cement solid.
Wet solid	Spent zeolite slurry. 14.2 w/o hydrated zeolite	B	590	cement solid.
Post Solidification D/D(e) Wastes:				
Trash	Cloth, paper, plastic, etc.	A	100	3:1 compaction
		B	31	3:1 compaction
		C(d)	69	3:1 compaction
Equipment and hardware	Predominantly steel, stainless steel, and Inconel	B	1,100	none
		C	110	none
		C(d)	180	none
Fuel storage racks	Aluminum alloy	B	620	none
		B	62	none
Rubble	Predominantly concrete	A	810	none
		B	81	none
		C(d)	65	none
Liquid	Spent decontamination solutions and secondary streams from the RTS(c)	C	600	cement solid.
		C	60	cement solid.
		C(d)	44	cement solid.
Wet solids	Resins and other slurries	C	700	cement solid.

(a) LLWTF: existing plant low-level waste treatment facility.
(b) FRS: Existing fuel receiving and storage pool.
(c) RTS: Planned radwaste treatment system.
(d) Denotes waste of a type having high probability of resulting in additional volumes that have a transuranic content exceeding 100 nCi/gm.
(e) D/D: Decontamination and decommissioning.

manageable number of waste streams. The annual volumes projected to be generated will also have to be addressed.*

The waste streams that may include volumes that exceed Class C concentrations are identified and the total volumes estimated using Table B.6 of the DOE FEIS (Ref. 75) as a guide. These are assumed to be trash, failed equipment, rubble, and D/D liquids as indicated in Table A-40, and are also assumed to exceed Class C limits primarily based on transuranic content.

The transuranic concentrations for such waste streams are conservatively estimated. A number of similar waste streams are also consolidated and the radionuclide concentrations given as a distribution (similar to the approach used for LWR process waste). For example, existing LLWTF sludge is combined with LLWTF resin, and FRS precoat sludge is combined with RTS filter backwash slurry. In addition, trash waste streams from previtrification decontamination, vitrification, and D/D are combined. Failed equipment streams are similarly combined. This consolidation is possible when two or more waste streams have similar isotopic (as opposed to concentration) distributions.

A summary of the 25 resulting consolidated waste streams is provided as Tables A-41 and A-42. These tables present a short description of the waste streams, the waste stream abbreviations, and the as-generated waste volumes projected to be generated each year. By "as-generated," the waste stream volumes and radionuclide concentrations are given prior to waste treatment procedures such as compaction for trash waste streams (VRF=3) and cement solidification for liquids and other wet waste streams (VIF=1.4). This is to allow comparison of different waste treatment options pursuant to the different waste spectra (see Section A.1.1 and Appendix B).

Finally, it should again be noted that the tabulated projections may vary considerably depending upon implementation of processing alternatives and experience.

A.7.4 Three Mile Island Decontamination

Decontamination activities at the damaged reactor have continued for several years. During this time, a variety of wastes have been generated, and more will be generated in the future. Much of the waste is physically and radiologically similar to waste routinely generated by normal operations at a nuclear power plant (trash, resins, etc.), although differences have been noted in the isotopic distributions. Other wastes which have and will be recovered, such as crumbled bits and pieces of nuclear fuel rods, are quite different than wastes generated from normal operations. The radionuclide concentrations within these latter wastes have frequently been at levels which were at the upper boundary or exceeded those in wastes normally generated from reactor operations.

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*For purposes of this report, projected as-generated volumes and concentrations of the Purex and Thorex HLW streams are included in the data base. This is done solely for comparison purposes. For example, it allows the user of the analysis methodology the opportunity to make some illustrative comparisons regarding the comparative hazards associated with dosposal of different types of waste. In reality, HLW generated during the project will be stored pending disposal into a geologic repository.

Table A-41. Summary of West Valley Demonstration Project Waste Streams

Symbol	Description
W-THORHLW	Thorex high-level waste (HLW)
W-PUREHLW	Purex HLW
W-COTRASH	Trash from existing facility systems
W-NCSOLID	Miscellaneous noncompressible dry solids
W-LLWTFRE	Low-level waste treatment facility (LLWTF) sludge and resins
W-FRSRESN	Fuel receiving and storage (FRS) facility filter precoat and resins
W-FRSLIQD	Radwaste treatment system (RTS) liquid waste
W-RTSRESN	RTS filter backwash and resins
W-LTTRASH	Compressible trash, low transuranic (TRU) content
W-HTTRASH	Compressible trash, high TRU content
W-LTEQUIP	Equipment and hardware, low TRU
W-HTEQUIP	Equipment and hardware, high TRU
W-PDWLIQD	Presolidification decontamination liquid waste
W-VITSUPR	Vitrification supernate liquids
W-VITWASH	Vitrification sludge wash liquids
W-VITSCRB	Vitrification scrub condensate liquids
W-VITMELT	Vitrification melter feed condensate liquids
W-VITFRAC	Vitrification fractionator condensate liquids
W-VITZEOL	Vitrification zeolite slurry
W-DDRACKS	Decontamination and decommissioning (D/D) fuel storage racks
W-DDLTRUB	D/D building rubble, low TRU content
W-DDHTRUB	D/D building rubble, high TRU content
W-DDLTLQD	D/D liquid, low TRU content
W-DDHTLQD	D/D liquid, high TRU content
W-DDRESIN	D/D ion exchange resins

Table A-42. Summary of West Valley Demonstration Project As-Generated Waste Streams and Volumes

Waste Description	As-Generated Volume/yr (m^3/yr)***																
	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000
Thorex HLW					15.7	15.7	15.7										
Purex HLW					666.7	666.7	666.7										
Trash from existing systems	771	771	771	771	771	771	771										
Miscellaneous dry solids	814	814	814	814	814	814	814										
LLWTF sludge and resins	66.9	66.9	66.9	66.9	66.9	66.9	66.9										
FRS filter precoat & resins	16.1	16.1															
RTS liquid waste					5.9	5.9	5.9										
RTS filter backwash & resins					7.4	7.4	7.4										
Trash, low TRU content	99	99			182.1	182.1	182.1			49.2	49.2	49.2	49.2	49.2	49.2	49.2	49.2
Trash, high TRU content	69.6	69.6			130.2	130.2	130.2			37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5
Equip. & hardware, low TRU	605	605			14	14	14			151.3	151.3	151.3	151.3	151.3	151.3	151.3	151.3
Equip. & hardware, high TRU	214.2	214.2			5.3	5.3	5.3			92.4	92.4	92.4	92.4	92.4	92.4	92.4	92.4
PD liquid waste	32.9	32.9															
Vit. supernate					500	500	500										
Vit. sludge wash					30.9	30.9	30.9										
Vit scrub condensate					188.1	188.1	188.1										
Vit. melter feed overheads					2.6	2.6	2.6										
Vit. frac. condensate					0.5	0.5	0.5										
Vit. zeolite slurry					140.5	140.5	140.5										
D/D fuel storage racks										85.3	85.3	85.3	85.3	85.3	85.3	85.3	85.3
D/D rubble, low TRU										111.4	111.4	111.4	111.4	111.4	111.4	111.4	111.4
D/D rubble, high TRU										9.2	9.2	9.2	9.2	9.2	9.2	9.2	9.2
D/D liquid, low TRU										58.9	58.9	58.9	58.9	58.9	58.9	58.9	58.9
D/D liquid, high TRU										20	20	20	20	20	20	20	20
D/D resins										62.5	62.5	62.5	62.5	62.5	62.5	62.5	62.5

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At the start of decontamination activities, it was recognized that much of the waste that would be generated would be of a physical form, radionuclide content, and/or radionucide concentration that was unusual compared with normal reactor operations. Given this, a memorandum of understanding was signed between NRC and DOE whereby unusual or "abnormal" waste generated from facility decontamination would be transferred to DOE for research and study (Ref. 77). This has been generally interpreted as wastes exceeding Class C concentrations as per 10 CFR Part 61. (Under normal power reactor operations, very little waste generated exceeds Class C concentrations.) This situation is expected to continue in the future.

Thus, most of the lower activity waste is being disposed at commercial disposal facilities. Generation of this waste has been approximated as discussed in Section A.2.2.1 through the assumption of a fictitious electrical power generation of 100 MW(e). Waste generation is then assumed to be proportional to the electrical capacity as discussed in Section A.2.2.2. The higher activity (abnormal) waste exceeding Class C concentrations has to date not been disposed at commercial disposal facilities, but has been transferred to DOE. This situation is assumed to continue in this report.

A.8 OTHER WASTE

For the purposes of this report, the characteristics of spent fuel are obtained from reference 59. Figure A.8 (Ref. 59) provides an illustration of the overall appearance of the fuel assemblies plus appropriate structural data.

Projections of spent fuel are made in a somewhat crude manner using information contained in reference 78, projections of nuclear power reactor electrical capacity presented in Section A.2.2, and projections of nuclear power reactor decommissioning schedules presented in Section A.7.2. Reference 78 contains a list of spent fuel stored as of 12/31/79 in nuclear power storage pools as a function of reactor type. Also indicated in this reference is the fuel stored at the GE reprocessing facility at Morris, Illinois, and the fuel formerly stored at the NFS reprocessing facility at West Valley, New York. The fuel stored at West Valley has since been returned to the generator. These data have been manipulated to arrive at the total fuel mass stored as a function of reactor type and NRC region. This information is listed below in units of metric tons of heavy metal (MTHM).

MTHM Stored as of 12/31/79

Reactor	Region 1	Region 2	Region 3	Region 4	Region 5	Total
PWR	890.9	798.2	666.8	140.7	100.8	2597.4
BWR	1033.9	395.7	1006.2	55.4	20.1	2511.3
Total	1924.8	1193.9	1673.0	196.1	120.9	5108.7

Beginning with the year 1980, additional fuel is assumed to be accumulated annually in accordance with the following simple equation, which estimates the annual fuel discharged for a typical large (1000 MW(e)) plant:

$$A = \frac{D \times C_f \times 10^3}{E_f \times B} \qquad \text{(A-8)}$$

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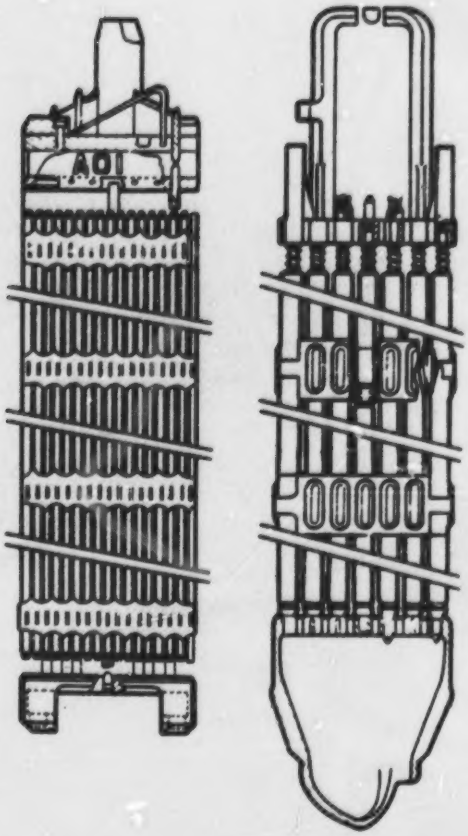

PWR BWR

Figure A.8. Typical Spent Fuel Assemblies

where:

A = MTHM discharged per Gigawatt-electric year;
D = days in the cycle;
C_f = plant capacity or plant load factor;
E_f = net electrical conversion efficiency of the plant; and
B = burnup in megawatt-days (thermal)/MTHM.

Typical values for these parameters are:

D = 365.25 days;
C_f = 0.7;
E_f = 0.32; and
B = 30,000 MWD/MTHM for a PWR
= 27,500 MWD/MTHM for a BWR

These assumptions result in an A value of about 26.6 MTHM/yr for a "typical" 1000 MW(e) PWR (or 0.0266 MTHM/MW(e)) and about 29 MTHM for a "typical" 1000 MW(e) BWR (0.029 MTHM/MW(e)). (Another estimate in reference 59 is 0.035 MTHM/MW(e)-yr for a PWR and 0.042 MTHM/MW(e)-yr for a BWR.) A "typical" PWR fuel assembly contains about 460 Kg of heavy metal so that about 58 fuel assemblies would be discharged per year per large reactor. A "typical" BWR fuel assembly contains about 183 Kg of heavy metal so that about 159 fuel assemblies would be discharged per year per large reactor (Ref. 78).

Another reference provides a list of physical characteristics of "typical" LWR fuel assemblies, and this list is included as Table A-43 (Ref. 67). Additional information on the physical characteristics of a number of fuel assembly designs is contained in reference 79.

Based on this table, one can arrive at a rough volume estimate for spent fuel generation. Based on overall outside dimensions of 0.0864 m^3 and 0.186 m^3 for BWR and PWR fuel assemblies, respectively, the annual gross volume generation rate comes to about 0.029 x 0.186/0.4614 = 0.012 m^3/MW(e) for a PWR and 0.013 m^3/MW(e) for a BWR. This also implies a stored spent fuel volume to 12/31/79 of:

m^3 stored as of 12/31/79

Reactor	Region 1	Region 2	Region 3	Region 4	Region 5	Total
PWR	359.1	371.8	268.8	56.7	40.6	1097.0
BWR	487.3	186.5	474.3	26.1	9.5	1183.7
Total	846.4	558.3	743.1	82.8	50.1	2280.7

This is not the whole story, however, since each spent fuel assembly is composed of a number of individual fuel rods arranged in a rectangular array, and held in place by spacer grids, tie rods, metal end fittings, and other miscellaneous hardware. One option under consideration (but not yet implemented aside from a few test assemblies) for long-term waste storage and eventual disposal is to

Table A-43. Typical Physical Characteristics of LWR Fuel Assemblies[a,b]

	BWR	PWR
Overall assembly length, m	4.470	4.059
Cross section, cm	13.9 x 13.9	21.4 x 21.4
Fuel element length, m	4.064	3.851
Active fuel height, m	3.759	3.658
Fuel element OD, cm	1.252	0.950
Fuel element array	8 x 8	17 x 17
Fuel elements per assembly	63	264
Assembly total weight, kg	319.9	657.9
Uranium/assembly, kg	183.3	461.4
UO_2/assembly, kg	208.0	523.4
Zircaloy/assembly, kg	99.5[c]	108.4[d]
Hardware/assembly, kg	12.4[e]	26.1[f]
Total metal/assembly, kg	111.9	134.5
Nominal volume/assembly, m^3	0.0864[g]	0.186[g]

[a] Ref. 67.

[b] Characteristics are based on 1976-1977 General Electric design data and 1975-1977 Westinghouse design data.

[c] Includes Zircaloy fuel-element spacers and fuel channel.

[d] Includes Zircaloy control-rod guide thimbles.

[e] Includes stainless steel tie-plates, Inconel springs, and plenum springs.

[f] Includes stainless steel nozzles and Inconel-718 grids.

[g] Based on overall outside dimension.

remove the hardware from the fuel rods. This allows the fuel rods, which contain the fission products which are of primary interest in terms of geologic repository disposal, to be consolidated into a smaller volume. This enables more economical storage and easier handling for transport and disposal. The hardware, which is composed of various types of corrosion-resistant metals such as Inconel or Zircaloy, becomes a second waste stream which could potentially be safely disposed by a less expensive method than a geologic repository.

For the purposes of this report, two separate waste streams are projected. These waste streams are included in this report solely to provide some points of comparison with impacts estimated for other waste streams. These waste streams are summarized below and consist of (1) spent fuel hardware, and (2) consolidated fuel rods.

Symbol	Description
L-FUEHARD	Spent fuel assembly hardware
L-SPENTFU	Consolidated spent fuel rods

L-FUEHARD. Depending upon parameters such as the fuel irradiation history and the hardware elemental composition, particular pieces of separated hardware may or may not exceed Class C concentrations. Based on Table A-43 about 56.6 kg of hardware would be generated per MTHM of PWR fuel, or about 67.6 kg of hardware per MTHM of BWR fuel. Hardware volumes are more difficult to estimate, since the hardware will be in a convoluted shape and will contain voids. Subtracting the volume comprised by the fuel element array by the nominal volume per assembly, one obtains a rough hardware volume of 7.9E-3 m^3 per BWR assembly, or 9.6E-3 m^3 per PWR assembly. This in turn implies a volume generation rate of 4.3E-2 m^3/MTHM for BWRs and 2.1E-2 m^3/MTHM for PWRs.

This may not be the whole volume, however, since there will be spacers and other materials in addition to the end pieces. (The spacers would be very easily compacted in any case.) DOE in reference 59 estimates 0.056 m^3/MTHM for fuel assembly hardware based on an assumed PWR to BWR ratio of 2:1. An overall density of 1 g/cm^3 is also assumed in this reference, which implies a mass generation rate for hardware of 56 kg/MTHM.

In this report, a somewhat high overall hardware volume generation rate of about 0.06 m^3/MTHM is assumed for LWR hardware. Due to the approximate nature of the projection, no differentiation is made between PWRs and BWRs. Using information provided above, this results in the following projected stored hardware volume as of 12/31/79:

m^3 Hardware as of 12/31/79

Region 1	Region 2	Region 3	Region 4	Region 5	Total
115.5	71.6	100.4	11.8	7.3	306.6

Volumes for 1980 and later years are estimated using the electrical capacity projections given in Table A-8 for PWRs and Table A-9 for BWRs. Volume projections as a function of reactor type are estimated as 0.0266 x 0.06 = 1.6E-3 m^3/MW(e)-yr for PWRs and 0.029 x 0.06 = 1.7E-3 m^3/MW(e)-yr for BWRs. Concentration projections are based on information from reference 59, or 1.49E+5 Ci/m^3. This is only a rough approximation, and actual concentrations would be expected to vary considerably.

L-SPENTFU. This waste stream consists of spent fuel rods resulting from consolidation activities. Waste package designs are currently under development by DOE, and will vary depending upon the characteristics of the disposal media. The wide variety in spent fuel characteristics (e.g., sizes and numbers of fuel rods per assembly, heat generation rates, fuel burnup) will also have considerable influence on the designs. Typical example conceptual designs are presented in references 80 and 81. Essentially, the fuel rods will be packed tightly together within a steel container which may or may not have an overpack.

The approach taken is to project volumes for fuel rods in an "untreated" form--i.e., the gross volumes for the spent fuel rods themselves while held within the hardware. After this, a rough volume reduction factor is assumed which represents consolidation within a container.

Based on Table A-43, the "untreated" volume of a "typical" PWR fuel assembly, less hardware, is about 0.21 x 0.21 x 3.85 = 0.18 m^3. That for a typical BWR is estimated to be 0.08 m^3. These volumes respectively correspond to about 0.39 m^3/MTHM for PWRs and 0.44 m^3/MTHM for BWRs, as well as a total stored volume (as of 12/31/79) of:

m^3 as of 12/31/79

Reactor	Region 1	Region 2	Region 3	Region 4	Region 5	Total
PWR	347.5	311.3	260.1	54.9	39.3	1013.0
BWR	454.9	174.1	442.7	24.4	8.8	1050.0
Total	802.4	485.4	702.8	79.3	48.1	2063.0

Volumes of 1980 and later years are estimated using the electrical capacity projections given in Tables A-8 and A-9 for PWRs and BWRs. Volume projections as a function of reactor type are estimated as 1.0E-2 m^3/MW(e)-yr for PWRs and 1.3E-2 m^3/MW(e)-yr for BWRs. Total gross concentrations, including short-lived radionuclides, are estimated at 1.5E+6 Ci/MTHM.

The effects of packaging may be approximated by a volume reduction factor to account for packing within a container, and a volume increase factor due to waste container internal voids, wall thicknesses, and overpack (if any). An approximate volume reduction factor of 2.0 is assumed; however, a volume increase factor of 1.0 is assumed due to lack of information on waste container characteristics.

A.9 AS-GENERATED (UNTREATED) RADIONUCLIDE CONCENTRATIONS

This chapter presents the untreated radionuclide concentrations assumed for each of the waste streams considered in this report. The radionuclide concentrations given in this report are based primarily on those given in past reports using previous versions of the analysis methodology (Refs. 1, 4), although in many cases some minor changes have been made to reflect updated information. In such cases, the principal changes have generally been to the gross radionuclide concentrations, and the original radionuclide distributions have been retained but scaled to the new gross concentrations. This is particularly true for some of the LWR process wastes. As discussed in more detail in Section A.9.2, time and resources does not permit a more complete review of the radionuclide isotopic spectrum for these streams.

The first section in this chapter presents some introductory material. The remaining sections in this chapter provide untreated concentrations for wastes from various sources (nuclear power plants, institutional facilities, etc.) in the same order as that for Chapters A.2 through A.8.

A.9.1 Introduction

This introductory section briefly reviews three concepts important for the use of the waste stream data contained in this appendix. These concepts include (1) volume reduction and increase considerations, (2) average vs. distributed concentrations, and (3) short- vs. long-lived radionuclide concentrations.

Waste Volume Reduction and Increase Considerations

The radionuclide concentrations are given for what has been somewhat loosely termed untreated waste streams--that is, in an as-generated form prior to final processing and/or packaging. For example, concentrations for compactible trash streams are generally given for a waste form prior to possible compaction or incineration treatment processes. Concentrations for ion-exchange resins are generally given for a dewatered waste form, while concentrations for liquids are given for an unsolidified form.

This is done to enable calculation of radionuclide concentrations on a convenient basis for each of the possible processing options considered in this report. For each waste stream and each waste spectra, actual radionuclide concentrations are determined by multiplying the untreated concentrations by a pair of factors which account for the increase and/or decrease in waste volume that may result from waste processing or packaging. That is, the actual radionuclide concentration for a given waste stream and spectrum is given as follows:

$$C_s = C_u * (VRF/VIF) \tag{A-9}$$

where:

C_s = radionuclide concentration for a given waste stream and waste spectrum;

C_u = radionuclide concentration for a given untreated waste stream;

VRF = volume reduction factor for a processing and/or packaging option associated with a given waste stream and waste spectrum; and

VIF = volume increase factor for a processing and/or packaging option associated with a given waste stream and waste spectrum.

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As an example, assume that the radionuclide concentration for a particular trash waste stream is 1 Ci/m^3, and that this waste stream is incinerated and the resulting ashes solidified in cement in a 55-gallon drum. Assuming that incineration reduces the trash volume by a factor of 20, and the solidification process increases the ash volume by a factor of 1.4, the final as-shipped radionuclide concentration is $1*(20/1.4) = 14.3$ Ci/m^3.

VRF and VIF factors for each waste stream and each waste spectra are given in Appendix B.

Average vs. Distributed Waste Stream Radionuclide Concentrations

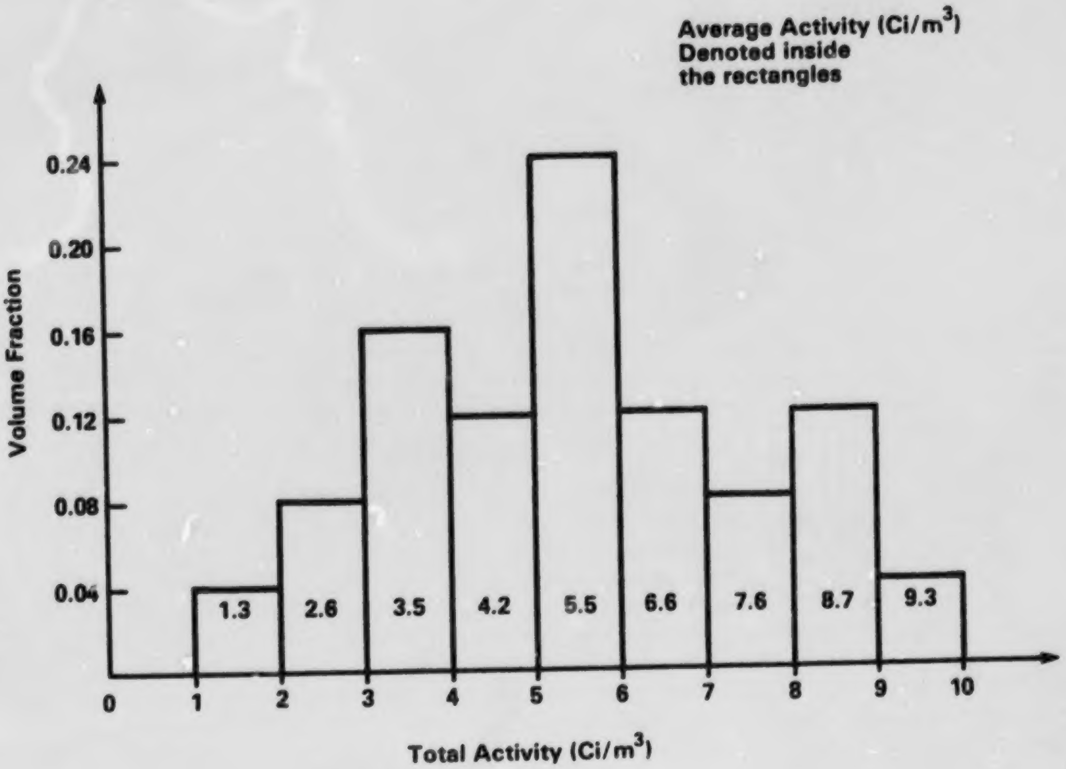
In the previous eight chapters frequent reference has been made to "average" vs. "distributed" gross radionuclide concentrations for individual waste streams. In the former case, a fixed average gross concentration is assumed across the waste stream, and concentrations of individual radionuclides are assigned to the waste stream based on this average gross concentration. This makes determining the classification status of the waste stream relatively simple; however, only the average classification status is determined, and frequently a waste stream may exist in a wide range of concentrations, with a corresponding range in classifications.

Given this, a procedure has been developed whereby gross radionuclide concentrations can be considered as a distribution across the waste stream rather than as an average. A series of concentration ranges are established, and within each range the percent of the waste stream volume that falls into the range is determined, as is the average gross concentration within the range. A fixed radionuclide spectrum is then assigned to each average gross concentration within each range. The concentrations of specific radionuclides thus vary according to the particular range considered, although the relative concentration ratio between two individual radionuclides is fixed for all concentration ranges.

To illustrate the data needed to determine concentration distributions, as well as to illustrate how the concentration distributions are stored and used in the codes, a hypothetical example distribution is presented in Figure A.9. This concentration distribution is typical of many observed distributions in that it somewhat resembles a rough Gaussian format; relatively small volumes of waste are found at the extreme high and low ends of the distribution, while an aggregation of waste volumes is seen at the middle of the distribution.

For this hypothetical example, data are assumed to have been collected which show that the gross concentration of the waste stream ranges from 1 to 10 Ci/m^3. Based upon this data, the range of possible concentrations has been divided into 10 increments--e.g., 1-2 Ci/m^3, 2-3 Ci/m^3, etc. For each increment the fraction of the total waste volume which has an concentration falling within the increment is determined, as well as the average concentration of the waste within the increment. For example, 24% of the waste is shown to fall within an increment of between 5 and 6 Ci/m^3 and to have an average concentration of 5.5 Ci/m^3.

Based upon these data, three hypothetical one-dimensional arrays can be specified which contain the following information:



ARRAYS

CL	0	1	2	3	4	5	6	7	8	9	10
PV	0	0	.04	.12	.28	.40	.64	.76	.84	.96	1.0
AA	0	0	1.3	2.17	2.93	3.31	4.13	4.52	4.81	5.30	5.46

Figure A.9. Example Total Activity Distribution

(1) The array CL, which contains for each increment the upper value of the concentration within the increment;

(2) The array PV, which contains for each increment the cumulative fraction of the waste which has concentrations less than the upper value of the particular increment; and

(3) The array AA, which contains for each increment the average concentration across the cumulative fraction of waste defined by the array PV.

These three arrays, and the values assigned to the arrays according to the example, are shown in Figure A.9.

The fact that the gross concentration of some of the waste streams is given as a distribution rather than as an average over the stream complicates determining the waste classification status of a waste stream. This results from the likelihood that a certain fraction of a distributed waste stream may be determined to be Class A, another fraction may be determined to be Class B, and so forth. In addition, each time that a fraction of a waste stream is classified, the concentration distribution across the remaining fraction must be recalculated.

Briefly, the classification test procedure for distributed waste streams works as follows:

(1) obtain an upper total concentration value by comparing to a set of radionuclide concentrations which will permit disposal of the waste in the waste class for which it is being considered,

(2) based on the array CL defined above, identify the fraction of the waste acceptable in this class (interpolating where necessary),

(3) take out the fraction which has been found to be acceptable from the distribution and redefine the waste distributions, and

(4) reconsider the new distribution for further classification.

Further details on the manner in which the distribution procedure is used in the codes are provided in reference 4.

Short- vs. Long-Lived Radionuclide Concentrations

In this report, the principal emphasis has been on estimating and comparing impacts over the long term. Thus, the half-lives of the 53 radionuclides considered in this report are generally measured in terms of years--frequently thousands or millions of years. However, much of the data that has been collected on concentrations of individual waste streams has included short-lived as well as longer-lived radionuclides. In order not to lose the data that has been collected, these short-lived radionuclides are generally included in the following sections. Radionuclides that are not considered in detail in the calculations are included as an "other" group of nuclides. This is done in order to retain the relationships between as-shipped concentrations, which include the shorter-lived nuclides, and the concentrations of longer-lived radionuclides.

A.9.2 Nuclear Power Plants

Radionuclide concentrations for wastes from nuclear power plants are given for three different groups of waste streams. These include process and trash waste streams for which radionuclide concentrations are given as a distribution, cartridge filters for which concentrations are given as an average, and other waste streams. Radionuclide concentrations are assumed based on data obtained from reference 1 as adjusted by new data discussed in this section and in Chapter A.2. The isotopic distributions, however, remain the same.

The isotopic distributions for reference 1 were made using a limited amount of data. Since that time, considerable additional data is in the process of being obtained based on waste sampling programs carried out by NRC, utilities, and others. It would be very useful to collect this additional data, interpret it, and develop updated isotopic distributions for the generic waste streams. However, this would require an effort that would exceed the time and monetary resources available for this project.

Distributed Process Waste Streams

Untreated radionuclide concentrations for the P-IXRESIN, P-CONCLIQ, P-FSLUDGE, B-IXRESIN, B-CONCLIQ, and B-FSLUDGE waste streams are given in Table A-44. These concentrations are scaled from reference 1 so that the radionuclide sum corresponds to the weighted average concentrations given in Table A-11. The gross concentrations for all these waste streams are all given as distributions across the waste stream volumes. These distributions are given in Table A-12.

Cartridge Filters and Trash

This group of waste streams includes PWR filter cartridges (P-FCARTRG) and LWR combustible and noncombustible trash (P-COTRASH, P-NCTRASH, B-COTRASH, and B-NCTRASH). Radionuclide concentrations for trash waste streams are given as scaled to weighted average gross concentrations. Gross concentrations are given as distributions as listed in Table A-12. Radionuclide concentrations for the P-FCARTRG waste stream are given as an average across the entire waste stream rather than as a volume distribution. Concentrations for cartridge filters and trash waste streams are given in Table A-45.

Other LWR Waste Streams

This group of waste streams includes activated reactor core components (L-NFRCOMP) and waste from periodic decontamination of the primary coolant system of nuclear power plants (L-DECONRS).

The L-NFRCOMP waste stream concentration is given as a distribution (see Section A.4.2), and individual radionuclide concentrations are scaled from the weighted average gross concentration across the entire projected waste volume. To determine the fractional contribution of each radionuclide of concern, a survey has been made of disposal facility shipment records for 1984. The survey indicated 44 clearly identifiable shipments of activated metal components for disposal, including 26 shipments (715 ft^3, 114,000 Ci) from PWRs, 17 shipments (472 ft^3, 171,000 Ci) from BWRs, and 1 shipment from a university (7 ft^3, 4,650 Ci). The activity distributions for the principal reported radionuclides are summarized in Table A-46, along with the radionuclide activity distribution

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Table A-44. Weighted Average Radionuclide Concentrations For Distributed Process Waste Streams (Ci/m^3)

Radionuclide	P-IXRESIN	P-CONCLIQ	P-FSLUDGE	B-IXRESIN	B-CONCLIQ	B-FSLUDGE
H-3	6.13E-1	2.66E-2	1.89E-2	2.32E-2	1.95E-3	1.35E-2
C-14	2.25E-2	9.80E-4	6.97E-4	1.44E-3	1.22E-4	8.32E-4
Fe-55	5.39E-1	1.75E-1	2.26E+0	1.15E+0	2.38E-1	1.54E+0
Ni-59	6.43E-4	2.09E-4	2.71E-3	1.18E-3	2.45E-4	1.59E-3
Co-60	1.04E+0	3.40E-1	4.38E+0	1.92E+0	3.97E-1	2.58E+0
Ni-63	1.98E-1	6.45E-2	8.33E-1	2.60E-2	5.38E-3	3.47E-2
Nb-94	2.04E-5	6.62E-6	8.55E-5	3.74E-5	7.75E-6	5.02E-5
Sr-90	4.47E-2	1.94E-3	1.38E-3	4.40E-3	3.69E-4	2.53E-3
Tc-99	1.90E-4	8.26E-6	5.86E-6	9.25E-5	7.82E-6	5.35E-5
I-129	5.62E-4	2.44E-5	1.73E-5	2.47E-4	2.08E-5	1.42E-4
Cs-135	1.90E-4	8.26E-6	5.86E-6	9.25E-5	7.82E-6	5.35E-5
Cs-137	5.05E+0	2.20E-1	1.56E-1	2.47E+0	2.08E-1	1.42E+0
U-235	1.09E-5	4.75E-7	1.07E-6	6.44E-8	1.08E-7	3.55E-7
U-238	8.55E-5	3.73E-6	8.40E-6	5.08E-7	8.47E-7	2.79E-6
Np-237	2.09E-9	9.10E-11	2.05E-10	1.23E-11	2.07E-11	6.82E-11
Pu-238	5.99E-3	3.95E-4	3.48E-4	1.01E-4	6.22E-4	4.98E-4
Pu-239/40	4.20E-3	2.55E-4	1.13E-3	6.46E-5	2.95E-4	2.52E-4
Pu-241	1.83E-1	1.11E-2	4.93E-2	3.14E-3	1.44E-2	1.23E-2
Pu-242	9.20E-6	5.59E-7	2.48E-6	1.41E-7	6.14E-7	5.54E-7
Am-241	4.31E-3	2.31E-4	1.93E-3	2.80E-5	3.75E-4	1.67E-4
Am-243	2.90E-4	1.56E-5	1.30E-4	1.90E-6	2.53E-5	1.12E-5
Cm-243	2.29E-6	9.03E-8	2.26E-6	3.26E-8	8.10E-7	3.17E-7
Cm-244	3.18E-3	1.48E-4	1.29E-3	2.20E-5	6.41E-4	2.39E-4
Total:	7.71E+0	8.41E-1	7.71E+0	5.60E+0	8.68E-1	5.61E+0

Table A-45. Radionuclide Concentrations for PWR Cartridge Filters and LWR Trash Waste Streams (Ci/m^3)

Radionuclide	P-FCARTRG	P-COTRASH	P-NCTRASH	B-COTRASH	B-NCTRASH
H-3	2.77E-3	7.32E-4	5.03E-3	5.67E-5	2.46E-4
C-14	1.02E-4	2.70E-5	1.85E-4	3.50E-6	1.51E-5
Fe-55	1.34E+0	1.44E-2	9.86E-2	5.03E-3	2.18E-2
Ni-59	1.59E-3	1.72E-5	1.18E-4	5.21E-6	2.26E-5
Co-60	2.58E+0	2.78E-2	1.91E-1	8.47E-3	3.65E-2
Ni-63	4.91E-1	5.29E-3	3.63E-2	1.14E-4	4.92E-4
Nb-94	5.03E-5	5.45E-7	3.73E-6	1.64E-7	7.11E-7
Sr-90	2.02E-4	5.34E-5	3.68E-4	1.06E-5	4.63E-5
Tc-99	8.62E-7	2.28E-7	1.56E-6	2.25E-7	9.38E-7
I-129	2.55E-6	6.73E-7	4.61E-6	5.99E-7	2.59E-6
Cs-135	8.62E-7	2.28E-7	1.56E-6	2.25E-7	9.76E-7
Cs-137	2.30E-2	6.04E-3	4.15E-2	5.99E-3	2.59E-2
U-235	8.77E-7	1.91E-8	1.31E-7	1.02E-9	4.46E-9
U-238	6.91E-6	1.50E-7	1.03E-6	8.04E-9	3.49E-8
Np-237	1.69E-10	3.67E-12	2.52E-11	1.97E-13	8.50E-13
Pu-238	6.05E-4	1.44E-5	9.92E-5	1.92E-6	8.37E-6
Pu-239/40	9.15E-4	1.34E-5	9.14E-5	9.72E-7	4.19E-6
Pu-241	4.00E-2	5.82E-4	3.99E-3	4.72E-5	2.04E-4
Pu-242	2.01E-6	2.92E-8	2.01E-7	2.12E-9	9.21E-9
Am-241	3.95E-4	9.56E-6	6.55E-5	8.11E-7	3.52E-6
Am-243	2.65E-5	6.47E-9	4.43E-6	5.47E-8	2.36E-7
Cm-243	4.65E-7	6.63E-9	4.53E-8	1.62E-9	7.03E-9
Cm-244	2.65E-4	6.31E-6	4.31E-5	1.26E-6	5.43E-6
Total:	4.48E+0	5.50E-2	3.77E-1	1.97E-2	8.53E-2

Table A-46. Summary of Survey of Activated Metal Waste Shipments During 1984

Radionuclide*	Activity Distribution in Activated Metals (%)				
	PWR	BWR	University	Total	Ref. 1
H-3	1.39E-5	5.93	-	3.50	-
C-14	1.15E-5	3.38E-3	-	1.99E-3	6.41E-3
Cl-36	6.33E-4	-	-	2.48E-4	-
Cr-51	9.92E-2	-	-	3.88E-2	-
Mn-54	1.76	3.92	-	2.99	-
Fe-55	4.92E+1	4.35E+1	-	4.49E+1	5.52E+1
Fe-59	6.45E-3	-	-	2.52E-3	-
Co-58	1.96E-1	-	-	7.68E-2	-
Co-60	3.94E+1	4.43E+1	1.00E+2	4.29E+1	3.96E+1
Ni-59	5.78E-2	2.66E-1	-	1.79E-1	3.46E-2
Ni-63	9.40	2.94	-	5.42	5.17
Zn-65	1.81E-5	3.67E-4	-	2.25E-5	-
Sr-90	4.19E-4	6.94E-7	-	1.64E-4	-
Nb-94	9.36E-6	6.64E-5	-	4.25E-5	2.03E-4
Mo-93	3.33E-4	-	-	1.31E-4	-
Tc-99	1.18E-4	4.60E-5	-	7.35E-5	-
Cs-137	1.35E-3	2.98E-6	-	5.32E-4	-
Vol.(ft^3)	7.15E+2	4.72E+2	7.00	1.19E+3	5.10E+3**
Act.(Ci)	1.14E+5	1.71E+5	4.65E+3	2.91E+5	5.84E+5**
No. of ship.	26	17	1	44	-
Ave. Conc.(Ci/m^3)	5.63E+3	1.29E+4	4.65E+3	8.61E+3	4.04E+3

*Present in total quantity of at least 0.1 Ci.

**Approximation based on assumed shipment of 2,890 m^3 of activated metal waste over 20 years, or about 145 m^3/yr.

assumed in reference 1. Only those radionuclides present in quantities of at least 0.1 Ci (out of 291,000 Ci) are listed.

For longer-lived radionuclides, reasonable agreement seems to be made between shipping data and the assumptions in reference 1. Reference 1 appears to overestimate the Nb-94 and C-14 content and underestimate the Ni-59 content. In addition, shipment data indicates the presence of significant quantities of tritium (mostly from BWR control rods), a radionuclide having a 12.26-year half-life. Shipment data also indicates the presence of a number of other short-lived radionuclides such as Cr-51 or Co-58. Finally it may be noted that the radionuclide inventory observed in the study totaled 291,000 Ci, while the annual inventory suggested by reference 1 is on the order of 584,000 Ci/yr.

Assumed radionuclide concentrations for the L-NFRCOMP waste stream are listed below. Data obtained from the above survey are scaled to a weighted average concentration of 95.6 Ci/m³.

L-NFRCOMP

Radionuclide	Concentration (Ci/m³)
H-3	3.34
C-14	1.91E-3
Cl-36	2.38E-4
Cr-51	3.71E-2
Mn-54	2.85
Fe-55	4.29E+1
Fe-59	2.40E-3
Co-58	7.34E-2
Co-60	4.10E+1
Ni-59	1.71E-1
Ni-63	5.18
Zn-65	2.16E-4
Sr-90	1.57E-4
Nb-94	4.06E-5
Tc-99	7.03E-5
Cs-137	5.09E-4
	9.56E+1

Radionuclide concentrations for the L-DECONRS waste stream are given as a gross average over all the waste, and are obtained from reference 22 by scaling estimated curie quantities in Table 2.1 of reference 22 to the assumed waste stream gross concentration given in Section A.2.2.3. This is based on a conservative assumption that all of the radionuclides in the oxide layer at Dresden Unit 1 are removed by the decontamination process. The waste will contain large quantities of chelating agents. Concentrations are listed in Table A-47.

Table A-47. Estimated Average Radionuclide Concentrations (Ci/m^3) for L-DECONRS Waste Stream

Radionuclide	Concentration (Ci/m^3)
Ce-144/Pr-144	4.47E-1
Ce-141	<3.38E-4
Co-57	<3.38E-4
Co-58	6.95E-1
Co-60	1.89E+1
Eu-154	<3.76E-5
Fe-59	<7.52E-5
Fe-55	2.63E+0
Mn-54	2.48E-1
Ni-63	9.96E-1
Ru-103	1.13E-2
Ru-106/Rh-106	8.46E-1
Sb-124	<1.88E-3
Sb-125	<1.88E-3
Zr-95/Nb-95	1.50E-1
Pu-239/40	7.52E-3
Pu-238	1.13E-2
Cm-241	3.76E-3
Cm-242/43	1.13E-2
Cm-244	3.76E-3
Total:	<2.50E+1

A.9.3 Other Nuclear Fuel Cycle Facilities

Waste streams include those from (1) uranium conversion and fuel fabrication, and (2) plutonium facility decontamination and fuel burnup studies.

Uranium Conversion and Fuel Fabrication Waste

These waste streams (F-PROCESS, F-COTRASH, F-NCTRASH, and U-PROCESS) are believed to account for only a small fraction of the waste volume sent to low-level waste disposal facilities and are also believed to be radiologically insignificant by comparison. Given this, no change is assumed at this time for the radionuclide concentrations given in the draft Part 61 EIS (Ref. 3). These concentrations are given for U-235 and U-238 as averages across the entire waste streams as summarized below:

Waste Streams (Ci/m^3)

Radionuclide	F-PROCESS	F-COTRASH	F-NCTRASH	U-PROCESS
U-235	2.30E-5	1.18E-6	1.13E-6	1.65E-5
U-238	8.54E-5	4.40E-6	4.20E-6	3.64E-4
Total:	1.08E-4	5.58E-6	5.33E-6	3.81E-4

Plutonium Facility Decontamination and Fuel Burnup Studies

Two waste streams fall into this group: L-PUDECON and L-BURNUPS.

The L-PUDECON waste stream will be generated during decontamination of some former mixed oxide fuel fabrication facilities, and approximately 7,900 ft^3 of trash-like waste is projected to be generated having transuranic concentrations exceeding 100 nCi/gm. The facility generating the bulk of the waste volume processed a mixture of depleted uranium dioxide and plutonium dioxide with the depleted uranium dioxide comprising approximately 80% of the mass. The isotopic composition of the plutonium used in the facility is shown in Table A-48. As shown, the isotopic variation across the various plutonium sources is relatively small, at least for purposes of waste disposal (Ref. 33).

Table A-48. Isotopic Variation Across Sources of Plutonium (in percent)

Isotope	Sefor	Consumer Power	West Valley	Halden	Average
Pu-238	0.036	0.425	0.079	0.294	0.3
Pu-239	90.625	76.241	85.829	79.885	79.9
Pu-240	8.334	16.336	11.531	14.509	14.5
Pu-241	4.770	2.910	1.156	2.243	2.2
Pu-242	0.052	1.179	0.250	0.826	0.8
Am-241	0.467	2.857	1.135	2.203	2.2

Based on these data and those of references 34 and 35, the L-PUDECON waste stream is assumed to have an average alpha concentration of about 200 nCi/gm. Using the isotopic distributions in Table A-47 and scaling a 200 nCi/gm concentration to all radionuclides except Pu-241, this implies an isotopic distribution of the following in units of nCi/gm:

Radionuclide	Concentration (nCi/gm)
Pu-238	50.8
Pu-239/40	79.8
Pu-241	2407.
Pu-242	0.03
Pu-241	69.4
	2607.

Assuming a density of 1.6 g/cm^3, this results in a concentration of about 4.17 Ci/m^3.

The assumed radionuclide concentrations are listed in Table A-49.

Table A-49. Assumed Radionuclide Concentrations for L-PUDECON Waste Stream

Radionuclide	Concentration (Ci/m^3)
Pu-238	8.13E-2
Pu-239/40	1.28E-1
Pu-241	3.85E+0
Pu-242	4.80E-5
Am-241	1.11E-1
Total:	4.17E+0

L-BURNUPS. This waste stream approximates waste generated by three facilities. Data from one facility (see Section A.4.3) implies a gross transuranic concentration distribution as follows:

Volume Percent	TRU Concentration (nCi/gm)
65	9.0E+1
29	6.0E+5
6	1.5E+6
Weighted Ave:	2.64E+5

This distribution is applied to waste from all three facilities, although some variation among facilities would actually be expected. Based on information from the waste generators, for example, the transuranic concentration of the lowest activity portion of the waste (90 nCi/gm) would actually be expected to vary from about 10 to about 300 nCi/gm (Ref. 37).

This waste stream is thus given as a distributed waste stream. The assumed radionuclide concentration distribution for this waste stream is based on scaling the weighted average concentration given above to the fission spectrum estimated for "typical" light water reactor fuel as given in reference 59. This is given as a weighted average assuming that nuclear power plants exist in a capacity ratio of about 2 PWR: 1 BWR, and that the average fuel burnup is 29,300 MWd/MTHM.

The radionuclide spectrum assumed for the waste stream is given in Table A-50, and weighted so that the transuranic concentration (not including Pu-241 and Cm-242) corresponds to 449 Ci/m^3 (2.64E+5 nCi/gm). This assumes an average waste density of 1.7 gm/cm^3 (chosen since most of the high activity waste is solidified liquid or metal pieces). Other radionuclides in the waste stream must also be considered, and are scaled to the weighted average transuranic concentration. This corresponds to an overall gross concentration distribution for the L-BURNUPS waste stream of:

Volume Percent	Gross Concentration (Ci/m^3)
65	1.54E+1
29	1.03E+5
6	2.57E+5
Weighted Ave:	4.52E+4

The radionuclides considered in this distribution are those listed in Table A-50.

Table A-50. Radionuclide Concentration Distribution for L-BURNUPS Waste Stream

Radionuclide	Concentration (Ci/m^3)	Radionuclide	Concentration (Ci/m^3)
H-3	4.16E+1	U-234	2.08E-3
C-14	7.93E-2	U-235	1.59E-3
Fe-55	8.92E+0	U-236	2.18E-2
Co-60	9.91E+0	U-238	3.17E-2
Ni-59	2.97E-3	Np-237	3.07E-2
Ni-63	3.96E-1	Pu-236	2.28E-2
Sr-90	6.44E+3	Pu-238	2.08E+2
Tc-99	1.29E+0	Pu-239/40	7.33E+1
Ru-103	7.14E+0	Pu-241	1.09E+4
Ru-106	1.68E+4	Pu-242	1.59E-1
Ag-110m	6.54E+1	Am-241	3.67E+1
Sn-126	4.76E-2	Am-243	1.39E+0
I-129	3.27E-3	Cm-242	3.57E+2
Cs-135	2.68E-2	Cm-243	3.86E-1
Cs-137	9.12E+3	Cm-244	1.29E+2
Eu-152	1.19E+0		
Eu-154	5.45E+2	Total:	4.51E+4
Eu-155	3.87E+2		

A.9.4 Institutional Facilities

The waste streams assumed from institutional facilities include dry solids (I-COTRASH and I+COTRASH), scintillation liquids and vials (I-LIQSCVL and I+LIQSCVL), other mostly aqueous liquids (I-ABSLIQD and I+ABSLIQD), and biological waste (I-BIOWAST and I+BIOWAST). These are generally very low activity waste streams, and the radionuclide concentrations assumed in the draft Part 61 EIS (Ref. 3) are also assumed here. These radionuclide concentrations are listed in Table A-51. Concentrations for all waste streams are given as averages across the entire waste stream.

Table A-51. Average Radionuclide Concentrations for Institutional Waste Streams (Ci/m^3)

Radionuclide	I-COTRASH*	I-LIQSCVL**	I-ABSLIQD#	I-BIOWAST##
H-3	9.13E-2	5.01E-3	1.42E-1	1.75E-1
C-14	5.26E-3	2.51E-4	8.16E-3	1.01E-2
Co-60	1.04E-2		3.12E-2	3.99E-3
Sr-90	1.45E-3	4.34E-3	4.34E-3	8.33E-3
Tc-99	3.39E-9		1.02E-8	6.51E-9
Cs-137	4.56E-3		1.37E-2	8.76E-3
Am-241	4.82E-6			
Total:	1.13E-1	9.60E-3	1.99E-1	2.06E-1

*Concentrations for the I+COTRASH waste stream are identical.
**Concentrations for the I+LIQSCVL waste stream are identical.
#Concentrations for the I+ABSLIQD waste stream are identical.
##Concentrations for the I+BIOWAST waste stream are identical.

A.9.5 Industrial Facilities

A.9.5.1 Low Activity Waste Streams

These waste streams include source and special nuclear material trash (N-SSTRASH and N+SSTRASH), other miscellaneous source and special nuclear material waste (N-SSWASTE), low activity trash similar to institutional trash (N-LOTRASH and N+LOTRASH), and other low activity industrial waste (N-LOWASTE). Radionuclide concentrations for these waste streams are given as averages across the total waste streams as listed in Table A-52. These waste streams are believed to be of generally low activity and the radionuclide concentrations are therefore adopted without change from the draft Part 61 EIS (Ref. 3).

A.9.5.2 Higher Activity Waste Streams

The higher activity waste streams consist of medical isotope production waste, waste from large industrial manufacturers using tritium, waste from large sealed source manufacturers, waste from small tritium manufacturers and users, discarded sealed sources and devices, and activated metals. Considerable modifications to the data base have been made since the draft Part 61 EIS.

Table A-52. Average Radionuclide Concentrations for Industrial Lower Activity Waste Streams (Ci/m^3)

Radionuclide	N-SSTRASH*	N-SSWASTE	N-LOTRASH**	N-LOWASTE
H-3			2.85E-2	1.63E-2
C-14			1.64E-3	9.36E-4
Co-60			3.25E-3	1.47E-3
Sr-90			4.53E-4	1.31E-3
Tc-99			1.06E-9	7.76E-10
Cs-137			1.42E-3	1.04E-3
U-235	2.36E-6	4.60E-5		
U-238	8.08E-6	1.71E-4		
Am-241			1.51E-6	
Total:	1.12E-5	2.17E-4	3.53E-2	2.11E-2

*Radionuclide concentrations for the N+SSTRASH waste stream are identical.

**Radionuclide concentrations for the N+LOTRASH waste stream are identical.

Medical Isotope Production Waste

Two waste streams are considered here. One high activity waste stream, N-ISOPROD, represents waste arising from hot cell operations. A low activity waste stream, N-ISOTRSH, represents waste arising from balance-of-plant operations. To estimate the radionuclide concentrations, waste shipment records for the year 1983 were reviewed. Of 71 waste shipments (2,919 ft^3) made in 1983, an isotopic inventory was available for all but one shipment. Two shipments (1884 ft^3 and 3,855.16 mCi) were of very low activity while the remaining 61 shipments for which an isotopic inventory could be determined (1,020 ft^3 and 7,540,800 mCi) were of high activity. The total isotopic inventory listed for each of the two waste types is presented in Table A-53. In addition to the isotopic distributions, the shipment records total 424.86 grams of special nuclear material (SNM) for the low activity wastes and 6,853.74 grams of SNM for the high activity wastes.

Isotopic concentrations for the two waste streams are made using conservative assumptions, information obtained from the above shipment records, and scaling factors for transuranic and other radionuclides as presented in Appendix B of reference 1. Concentrations for the radionuclides listed in Table A-53 are calculated using the listed volumes. Concentrations for U-235 and U-238 are estimated using the SNM shipment data and an assumed enrichment of 93%. Concentrations for other nontransuranic radionuclides (H-3, C-14, Tc-99, I-129, and Cs-135) are estimated using the scaling factors presented in reference 1, and scaling to the total concentration of Cs-137 and Sr-90 in the waste stream of interest. These totaled 3.93 Ci/m^3 for N-ISOPROD and 6.0E-4 Ci/m^3 for N-ISOTRSH.

Concentrations for transuranic radionuclides are estimated using scaling factors from reference 1, and scaling to assumed total transuranic concentrations of 1.0 nCi/gm for N-ISOPROD and 1.0E-3 nCi/gm for N-ISOTRSH. The transuranic concentration for N-ISOPROD is consistent with the assumptions given in reference 1, Table A-53. Summary of Isotopic Inventories for Medical Isotope Production Waste Streams for 1983

Isotope	High Activity Waste (mCi)	Low Activity Waste (mCi)
Cr-51	106,136	
Mn-54	76,729	
Fe-55	466,116	
Co-58	176,676	
Fe-59	14,737	
Co-60		33.38
Ni-63	7,180	
Sr-89	575,623	
Sr-90	68,441	16.00
Y-91	906,137	
Zr-93		46.74
Nb-95	1,614,958	251.95
Zr-95	899,037	93.75
Mo-99		24.75
Ru-103	160,932	70.67
Ru-106	70,625	
Ag-110m		357.96
Sn-113		42.25
In-113m		42.25
Sb-124		4.38
I-125		2,396.7
Cs-137	45,005	16.00
Ba-140		202.9
La-140		254.55
Ce-141	216,919	
Ce-144	1,485,776	
Pm-147	253,685	
U-235		0.93
Volume (ft^3):	1,020	1,884

while the assumed transuranic concentration for N-ISOTRSH is conservatively estimated using the Cs-137/Sr-90 concentrations in the two waste streams as a guide. (A ratio of 10^3 is conservatively assumed, although the shipment records actually imply a ratio of 6,500.) Waste stream densities used in the calculations were 1.7 g/cm^3 for N-ISOPROD and 0.6 g/cm^3 for N-ISOTRSH.

Radionuclide concentrations were finally scaled to an average concentration of 245 Ci/m^3 for N-ISOPROD and 0.065 Ci/m^3 for N-ISOTRSH, and are listed in Table A-54.

Waste From Large Tritium and Carbon-14 Manufacturers

NE Region. These consist of nine waste streams of which six are of relatively low activity and three are of relatively high activity. As-generated radionuclide concentrations are presented in Table A-55. These concentrations are estimated principally based on inspection of shipment records.

MW Region. Radionuclides associated with this group of waste streams are principally tritium and carbon-14. Average radionuclide concentrations are listed in Table A-56 for waste in an as-generated form. Radionuclide estimates are made principally based on shipment records.

Sealed Source Manufacturing Waste

Limited data exist for these waste streams. For the N-SORMFG1 waste stream, the principal activity appears to be Am-241. About 60% of the volume appears to be of relatively low activity and is composed of contaminated metal scrap. The remaining 40% of the volume is of higher activity, consisting of americium oxide dissolved in a glass matrix. The approximate projected Am-241 concentrations are listed below:

Waste Volume (%)	Concentration (Ci/m^3)
60	5.0E-3
40	2.0E+1

For the N-SORMFG2 waste stream, the radionuclide contaminants are assumed to be Co-60 and Cs-137. The gross concentration distribution is assumed to be as follows:

Volume Percent	Concentration (Ci/m^3)
99	5.65E-2
1	3.51E+2
Weighted Ave:	3.57E+0

Table A-54. Radionuclide Concentrations (Ci/m^3) for N-ISOPROD and N-ISOTRSH Waste Streams

	Waste Stream Conc. (Ci/m^3)			Waste Stream Conc. (Ci/m^3)	
Radionuclide	N-ISOPROD	N-ISOTRSH	Radionuclide	N-ISOPROD	N-ISOTRSH
H-3	1.09E-2	1.51E-6	I-125	--	4.04E-2
C-14	1.17E-5	1.62E-9	I-129	7.04E-7	9.77E-11
Cr-51	3.64E+0	--	Cs-135	8.48E-5	1.18E-8
Mn-54	2.63E+0	--	Cs-137	1.54E+0	2.70E-4
Fe-55	1.60E+1	--	Ba-140	--	3.42E-3
Co-58	6.06E+0	--	La-140	--	4.29E-3
Fe-59	5.05E-1	--	Ce-141	7.44E+0	--
Co-60	--	5.63E-4	Ce-144	5.09E+1	--
Ni-63	2.46E-1	--	Pm-147	8.70E+0	--
Sr-89	1.97E+1	--	U-235	4.70E-4	3.00E-5
Sr-90	2.35E+0	2.70E-4	U-238	5.48E-6	1.67E-7
Y-91	3.11E+1	--	Np-237	1.03E-13	3.31E-17
Zr-93	--	7.88E-4	Pu-238	3.81E-5	1.22E-8
Nb-95	5.54E+1	4.25E-3	Pu-239/40	1.07E-5	3.44E-9
Zr-95	3.08E+1	1.58E-3	Pu-241	1.37E-3	4.40E-7
Mo-99	--	4.17E-4	Pu-242	1.85E-8	5.94E-12
Tc-99	8.48E-5	1.18E-8	Am-241	2.12E-6	6.80E-10
Ru-103	5.52E+0	1.19E-3	Am-243	2.43E-7	7.77E-11
Ru-106	2.42E+0	--	Cm-242	2.27E-4	7.29E-8
Ag-110m	--	6.03E-3	Cm-243	5.57E-8	1.79E-11
Sn-113	--	7.12E-4	Cm-244	3.20E-5	1.03E-8
In-113m	--	7.12E-4			
Sb-124	--	7.38E-5	Total:	2.45E+2	6.50E-2

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Table A-55. As-Generated Radionuclide Concentrations for Waste Streams From Large Northeast Region Manufacturers

		Concentration (Ci/m^3)			
Symbol	Waste Stream	H-3	C-14	S-35	Total
N-NECOTRA	Compactible trash	3.80E-1	4.07E-2	1.19E-1	0.54
N-NEABLIQ	Absorbed organic liquids	5.48E+1	8.43E-1	1.89E+1	74.55
N-NESOLIQ	Solidified aqueous liquid	2.60E+1	1.56E-1	8.57E-2	26.27
N-NEVIALS	Reject product vials	3.09E+1	7.37E-1	1.59E+1	47.49
N-NENCGLS	Noncompactible glass	1.99E+1	1.88E-2	3.57E-2	19.96
N-NEWOTAL	Noncompactible wood and metal	4.15E-1	1.51E-1	4.98E-2	0.61
N-NETRGAS	Tritium gas	4.62E+4	0	0	4.62E+4
N-NETRILI	Absorbed tritiated liquid	4.62E+4	0	0	4.62E+4
N-NECARLI	Absorbed C-14 liquid	0	4.11E+2	0	4.11E+2

Table A-56. As-Generated Average Radionuclide Concentrations for Waste Streams From Large Midwest Region Manufacturers

		Concentration (Ci/m^3)		
Symbol	Waste Stream	H-3	C-14	Total
N-MWTRASH	Laboratory trash	1.59E+0	2.48E-1	1.84E+0
N-MWABLIQ	Absorbed organic liquid	1.93E+2	1.40E+1	2.07E+2
N-MWSOLIQ	Solidified aqueous liquid	1.01E+3	0	1.01E+3
N-MWWASTE	Miscellaneous waste	6.69E+0	2.83E+1	3.50E+1

The weighted mean radionuclide concentrations, assuming a Co-60/Cs-137 ratio of 4.5, are:

Radionuclide	Concentration (Ci/m³)
Co-60	2.92
Cs-137	0.65
	3.57

The N-SORMFG3 and N-SORMFG4 waste streams are both generated by the same facility located in NRC Region III. The N-SORMFG3 waste stream is assumed to principally contain Cs-137, and is also assumed to be generated at an average concentration of about 6,000 Ci/m³ prior to packaging in lead lined drums. The lead lined drums result in a volume increase of about 2 (Ref. 49). Any further volume increase, such as might result from use of waste overpacks, is not considered.

The N-SORMFG4 waste stream is composed of miscellaneous materials from other plant operations. It is assumed to be generated at a rate of about 160 m³/yr. Radionuclide concentrations are estimated based on inspection of shipping records. These are listed (in an as-shipped form) in Table A-57.

Table A-57. Average Radionuclide Concentrations for N-SORMFG4 Waste Stream

Radionuclide	Concentration (Ci/m³)
H-3	1.70E-1
C-14	2.40E-4
Sc-46	4.08E-4
Cr-51	2.27E-5
Fe-59	1.36E-4
Sr-85	1.86E-4
Sr-90	2.63E-3
Nb-95	1.82E-5
Ag-110m	1.14E-3
In-111	9.09E-9
In-114m	1.14E-5
I-125	1.42E-2
Cs-137	4.31E-1
Ce-141	2.50E-5
Pm-147	1.91E-1
Yb-169	3.21E-2
Po-210	2.64E-1
Th-228	4.54E-9
Total:	1.11E+0

Waste From Small Tritium Manufacturers and Users

These six high-activity waste streams are assumed to consist primarily of tritium. For some waste streams, carbon-14 is also evident, although in significantly smaller relative concentrations. Another consideration is that the isotopic distribution between carbon-14 and tritium appears to be highly variable.

The six waste streams and weighted-average tritium concentrations are given below. The tritium concentration distributions for these six waste streams has been given in Section A.4.5 in Table A-18.

Waste Stream	Description	Weighted-Average Tritium Conc. (Ci/m^3)
N-TRIPLAT	Tritium contained in paint or as a plating	489
N-TRITGAS	Tritium disposed in a gaseous form	594
N-TRISCNT	Tritium contained in scintillation liquid	42.2
N-TRILIQD	Tritium contained in aqueous liquid	711
N-TRITRSH	Tritium contained in or contacting miscellaneous trash (e.g. plastic, paper, glass)	69.4
N-TRIFOIL	Tritium contained in or absorbed into metal (e.g., foils)	780

Sealed Sources and Devices

A different approach is used for this group of waste streams. Nine separate waste streams are considered as described below, and the particular isotope considered for each waste stream is self-explanatory.

Designation	Description
N-TRITSOR	Tritium (H-3) source
N-CARBSOR	Carbon-14 source
N-COBLSOR	Cobalt-60 source
N-NICKSOR	Nickel-63 source
N-STROSOR	Strontium-90 source
N-CEISOR	Cesium-137 source
N-PLU8SOR	Plutonium-238 source
N-PLU9SOR	Plutonium-239 source
N-AMERSOR	Americium-241 source
N-PUBESOR	Plutonium-238 neutron source
N-AMBESOR	Americium-241 neutron source

For each waste stream, the code inputs (on a regional basis) the annual number of waste sources per waste stream, plus the activity distribution for the waste stream. Activities are used instead of concentrations, since the volumes of particular sources do not vary significantly across the activity range. The average dimensions of sources exceeding 10 curies are a length of about 3 inches and a diameter of about 0.5 inches. This translates to a volume of about 0.6 cubic inches (9.65E-6 m^3) per source. Typically, the source will be doubly encapsulated in stainless steel, although this is frequently not the case for the older alpha sources. The integrity of the encapsulation may also be frequently open to question, since one of the principal reasons for disposal of a particular source is because it develops a leak. Some of the old alpha sources are also known to be quite fragile.

Activated Metals (N-HIGHACT)

Radionuclide concentrations are given as averages across the entire waste stream. No change is assumed from the concentrations listed in the draft Part 61 EIS (Ref. 3). Concentrations for this waste stream are listed below:

Radionuclide	Concentration (Ci/m^3)
C-14	1.32E-2
Fe-55	1.15E+2
Ni-59	6.56E-2
Co-60	8.48E+1
Ni-63	1.06E+1
Nb-94	4.47E-4
Total:	2.10E+2

A.9.6 Other Non-Fuel Cycle Sources

Waste streams from these sources include waste streams containing discrete quantities of radium as well as waste streams associated with U.S. Navy operations.

Radium-Contaminated Waste

The predominate radioisotope is of course Ra-226, although there may be varying contributions by radium daughters depending upon the design of a particular source. In most commercial uses, radium is used in equilibrium with several short-lived daughters, including radon-222 through polonium-214. Equilibrium is reached in about 30 days. The only significantly long-lived daughter (other than stable lead-206) is lead-210 (half-life of 20.4 y), which reaches equilibrium with radium-226 in about 100 years.

In this report, the radionuclide content of the radium sources is assumed to consist of Ra-226. Radium daughters occurring prior to lead-210 generally have half-lives on the order of minutes (an exception is Rn-222) and the radiological dose contributions of these short-lived daughters are included with that of radium-226. The ingrowth of the longer-lived lead-210 is calculated as a function of time. For most sources, the radium content is given as a distribution as presented in Table A-24. One gram of radium-226 is assumed to equal one curie.

For the radium resin waste (N-RARESIN), lead-210 (plus its daughters) is assumed to be in equilibrium with the radium-226 in groundwater. The water treatment process is assumed to have similar decontamination factors for both radionuclides. As discussed in Section A.6.2.1, the radium-226 concentration is assumed to be 35 mCi/m^3 for a dewatered condition.

Military (Navy) Waste

Individual radionuclide concentrations for these two waste streams are given in Table A-58. For the M-NAVYDRY waste stream, radionuclide distributions are taken from the P-COTRASH, but scaled to an average gross concentration of 0.02 Ci/m^3. The M-NAVYWET waste stream is given as a gross concentration distribution (94% at 0.2 Ci/m^3 and 6% at 10 Ci/m^3) having a weighted average concentration of 0.789 Ci/m^3. The isotopic distribution from the P-IXRESIN waste stream is used for the M-NAVYWET waste stream, but scaled to 0.789 Ci/m^3.

Both of these isotopic distributions are believed to be conservative, given the inclusion of several fission products within the radionuclide distributions. Compared with nuclear power plant waste, however, Navy waste would be expected to be dominated by activation products rather than fission products.

A.9.7 Non-Routine Waste

A.9.7.1 Uranium Fuel Recycle

Assumed radionuclide concentrations for these 23 waste streams are listed in Tables A-59 through A-61. The radionuclide concentrations have been obtained based upon information presented in references 4 and 59. Given the speculative nature of any assumed generation of such waste, as well as the uncertainties involved in the waste stream generation, the waste stream concentrations are all presented as averages. Spent fuel is assumed to be processed after a delay time of 1.5 years following removal from the reactor.

Table A-58. Radionuclide Concentrations for M-NAVYDRY and M-NAVYWET Waste Streams (Ci/m^3)

Radionculide	M-NAVYDRY	M-NAVYWET
H-3	2.67E-4	6.26E-2
C-14	9.82E-6	2.29E-3
Fe-55	5.24E-3	5.51E-2
Ni-59	6.24E-6	6.57E-5
Co-60	1.01E-2	1.07E-1
Ni-63	1.92E-3	2.03E-2
Nb-94	1.97E-7	2.08E-6
Sr-90	1.95E-5	4.57E-3
Tc-99	8.26E-8	1.94E-5
I-129	2.44E-7	5.73E-5
Cs-135	8.26E-8	1.94E-5
Cs-137	2.20E-3	5.16E-1
U-235	6.92E-9	1.11E-6
U-238	5.46E-8	8.74E-6
Np-237	1.33E-12	2.13E-10
Pu-238	5.24E-6	6.12E-4
Pu-239/40	4.85E-6	4.29E-4
Pu-241	2.11E-4	1.87E-2
Pu-242	1.06E-8	9.40E-7
Am-241	3.47E-6	4.40E-4
Am-243	2.34E-7	2.97E-5
Cm-243	2.40E-9	2.34E-7
Cm-244	2.29E-6	3.25E-4
Total:	2.00E-2(a)	7.89E-1(b)

(a) Concentrations given as averages across the waste stream.
(b) Concentrations given as weighted averages across the (distributed) waste stream.

Table A-59. Radionuclide Concentrations for Liquid High-Level Waste Stream (R-HLLWFRP)

Radionuclides	Concentration (Ci/m³)	Radionuclides	Concentration (Ci/m³)	Radionuclides	Concentration (Ci/m³)
H-3	5.60E+1	Ba-137m	1.45E+5	Am-242m	8.08E+1
C-14	9.33E-1	Ce-141	1.63E+1	Am-242	8.08E+1
Se-79	5.67E-1	Ce-144	4.00E+5	Am-243	7.88E+1
Kr-85	0	Pr-144	4.00E+5	Cm-242	1.65E+4
Rb-87	2.67E-5	Pm-147	1.15E+5	Cm-243	1.62E+1
Sr-89	6.83E+2	Pm-148m	7.33E+0	Cm-244	1.19E+4
Sr-90	1.02E+5	Pm-148	5.83E-1	Cm-245	3.03E+0
Y-90	1.02E+5	Sm-151	2.00E+3	Cm-246	5.68E-1
Y-91	2.17E+3	Eu-152	2.67E+1	Cm-247	2.98E-6
Zr-93	2.67E+0	Eu-154	1.02E+4	Cm-248	1.24E-5
Zr-95	5.67E+3	Eu-155	7.00E+3	Bk-249	2.23E-2
Nb-93m	4.00E-1	Gd-153	1.00E+1	Cf-249	1.65E-4
Nb-95m	1.22E+2	Tb-160	1.03E+1	Cf-250	5.73E-4
Nb-95	1.27E+4	Ho-166m	1.13E-3	Cf-251	5.78E-6
Tc-99	2.17E+1	Ac-227	1.61E-6	Cf-252	5.62E-4
Ru-103	1.23E+2	Th-228	8.83E-3		
Ru-106	3.17E+5	Th-230	2.13E-5	Total:	2.34E+6
Rh-103m	1.23E+2	Th-231	2.65E-2		
Rh-106	3.17E+5	Th-234	5.25E-1		
Pd-107	2.00E-1	Pa-231	1.78E-5		
Ag-110m	1.37E+3	Pa-233	6.67E-1		
Ag-110	1.83E+2	Pa-234m	5.25E-1		
Cd-113m	1.67E+1	Pa-234	5.25E-4		
Sn-119m	5.50E+0	U-232	9.93E-5		
Sn-121m	8.33E-4	U-233	3.25E-7		
Sn-123	6.33E+2	U-234	2.80E-3		
Sn-126	9.17E-1	U-235	1.33E-4		
Sb-124	1.17E+0	U-236	2.20E-3		
Sb-125	9.83E+3	U-237	3.65E-2		
Sb-126m	9.17E-1	U-238	2.62E-3		
Sb-126	9.00E-1	Np-239	7.88E+1		
Te-123m	2.83E-2	Pu-236	2.68E-3		
Te-129m	4.00E+3	Pu-238	4.62E+1		
Te-127m	7.17E+2	Pu-239	2.98E+0		
Te-127	7.17E+2	Pu-240	6.10E+0		
I-129	2.92E-4	Pu-241	1.46E+3		
Cs-134	2.00E+5	Pu-242	3.25E-2		
Cs-135	5.17E-1	Pu-243	1.49E-8		
Cs-137	1.55E+5	Am-241	1.18E+3		

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Table A-60. Radionuclide Concentrations for Other Fuel Reprocessing Plant Waste Streams

Radionuclide	Waste Stream Concentrations (Ci/m³) R-FUEHARD	R-HULLFRP	R-ILLWFRP	R-SILIGEL	R-MPCOTRH	R-MPCOTRL	R-MPNCTRA	R-DEGREXT	R-MPRESIN	R-SBRESIN
H-3		2.4E+2	3.8E+0							
C-14	8.9E-1	2.3E-1	5.1E-5		1.53E-6	4.7E-10	1.87E-6	7.0E-9	1.1E-5	
Fe-55	7.1E+4	3.4E+2								7.78E+0
Co-60	7.1E+4	3.8E+2								4.86E+0
Ni-59	5.4E+1	1.1E-1								
Ni-63	7.4E+3	1.5E+1								
Sr-90	1.8E-2	1.2E+2	5.6E+0		1.65E-1	5.1E-5	2.03E-1	7.6E-4	1.2E+0	6.21E-1
Nb-94	1.8E-3									
Tc-99	1.3E-1	2.4E-2	1.3E-4		3.61E-5	1.1E-8	4.37E-5	1.6E-7	2.6E-4	
I-129			9.6E-4					4.4E-4	1.4E-2	
Cs-135		5.8E-4	2.8E-5		8.54E-7	2.6E-10	1.05E-6	3.9E-9	6.2E-6	
Cs-137		1.8E+2	8.5E+0		2.52E-1	7.8E-5	3.10E-1	1.2E-3	1.9E+0	3.69E+1
U-234		6.4E-4	3.1E-3		9.31E-7	2.8E-10	1.13E-6	4.3E-5	6.8E-4	
U-235		3.0E-5	1.5E-4		4.37E-8	1.3E-11	5.33E-8	2.0E-6	3.2E-5	
U-236		4.9E-4	2.4E-3		7.14E-7	2.2E-10	8.67E-7	3.3E-5	5.2E-4	
U-238		6.0E-4	2.9E-3		8.77E-7	2.7E-10	1.07E-6	4.0E-5	6.4E-4	
Np-237		7.5E-4	3.6E-5		1.10E-6	3.3E-10	1.33E-6	5.0E-5	8.0E-6	
Pu-236		6.0E-4	2.9E-3	6.4E-6	8.77E-7	2.7E-10	1.07E-6	4.0E-3	6.4E-4	
Pu-238		1.0E+1	5.1E+1	1.1E-1	1.53E-2	4.6E-6	1.87E-2	6.9E+1	1.1E+1	
Pu-239		6.8E-1	3.3E+0	7.2E-3	9.88E-4	3.0E-7	1.20E-3	4.5E+0	7.2E-1	
Pu-240		1.4E+0	6.6E+0	1.5E-2	1.98E-3	6.1E-7	2.40E-3	9.1E+0	1.5E+0	
Pu-241		3.4E+2	1.6E+3	3.6E+0	4.94E-1	1.5E-4	6.00E-1	2.3E+3	3.6E+2	
Pu-242		7.3E-3	3.6E-2	7.8E-5	1.07E-5	3.3E-9	1.32E-5	4.9E-2	7.8E-3	
Am-241		1.3E+0	6.5E-2		1.97E-3	5.9E-7	1.32E-3	8.9E-2	1.4E-2	
Am-243		8.8E-2	4.5E-3		1.31E-4	3.9E-8	1.60E-4	5.9E-3	9.4E-4	
Cm-242		1.9E+1	9.0E-1		2.74E-2	8.3E-6	3.33E-2	1.2E+0	2.0E-1	
Cm-243		1.8E-2	8.8E-4		2.63E-5	8.1E-9	3.23E-5	1.2E+0	1.9E-4	
Cm-244		1.4E+1	6.6E-1		1.97E-2	6.0E-6	2.40E-2	9.0E-1	1.4E-1	
Total:	1.49E+5	1.66E+3	1.68E+3	3.73E+0	9.78E-1	2.99E-4	6.01E-1	2.38E+3	3.77E+2	5.02E+1

Table A-60. (continued)

Radionuclide	R-SBCOLIQ	R-SBCOTRA	R-SBNCTRA	R-UFFINES	R-UFK2MUD	R-UFCOTRA	R-UFNCTRA	R-PUCOTRA	R-PUNCTRA
	Waste Stream Concentrations (Ci/m³)								
H-3									
C-14									
Fe-55	1.08E+1	5.28E-3	2.36E-3						
Co-60	6.77E+0	3.31E-3	1.44E-3						
Ni-59									
Ni-63									
Sr-90	8.60E-1	4.30E-4	1.86E-4						
Nb-94									
Tc-99									
I-129									
Cs-135									
Cs-137	5.13E+1	2.54E-2	1.10E-2						
U-234				1.51E-2	6.3E-3	3.15E-3	1.87E-3		
U-235				7.09E-4	3.0E-4	1.48E-4	8.80E-5		
U-236				1.15E-2	4.8E-3	2.41E-3	1.43E-3		
U-238				1.42E-2	5.9E-3	2.97E-3	1.76E-3		
Np-237									
Pu-236				1.16E-5				1.65E-2	2.58E-3
Pu-238				1.92E-1				2.75E+2	4.28E+1
Pu-239				1.27E-2				1.80E+1	2.77E+0
Pu-240				2.64E-2				3.64E+1	5.62E+0
Pu-241				6.35E+0				8.48E+3	1.30E+3
Pu-242				1.38E-4				1.95E-1	3.04E-2
Am-241									
Am-243									
Cm-242									
Cm-243									
Cm-244									
Total:	4.28E+1	3.44E-2	1.50E-2	6.63E+0	1.73E-2	8.68E-3	5.15E-3	8.81E+3	1.35E+3

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Table A-61. Radionuclide Concentrations for Mixed Oxide Fuel Fabrication Facility Waste Streams

	Waste Stream Concentrations (Ci/m^3)		
Radionuclides	R-MOXCOTR	R-MOXNCTR	R-MOXSOLN
U-234	6.42E-4	1.10E-4	1.15E-4
U-235	2.58E-5	4.13E-6	4.07E-6
U-236	1.26E-7	2.09E-8	2.19E-8
U-237	2.58E-2	4.13E-3	4.47E-3
U-238	5.25E-4	8.80E-5	9.47E-5
Np-237	2.63E-7	4.40E-8	4.78E-8
Pu-236	1.50E-3	2.48E-4	2.70E-4
Pu-238	3.08E+1	5.23E+0	5.62E+0
Pu-239	2.17E+0	3.53E-1	3.85E-1
Pu-240	4.33E+0	7.15E-1	7.71E-1
Pu-241	9.75E+2	1.60E+2	1.76E+2
Pu-242	2.33E-2	3.85E-3	4.16E-3
Am-241	1.60E+0	2.64E-1	5.21E+1
Total:	1.01E+3	1.66E+2	2.35E+2

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A.9.7.2 Nuclear Power Plant Decommissioning

Individual radionuclide concentrations for these waste streams are very difficult to project. For this report, however, the gross waste activities as listed in Table A-31 are scaled to radionuclide distributions as summarized below:

BWR Waste Streams

Waste Stream	Description	Scaling Reference
B-DECORES	Activated core shroud	Table 5.2, Ref. 82
B-DEACINT	Activated reactor internals	Table 5.2, Ref. 82
B-DEACVES	Activated reactor Vessel	Table 5.3 Ref. 82
B-DEACTCO	Activated concrete	Tables 5.-5 & 5.7, Ref. 82
B-DECONME	Contaminated metal	Table B-27, Ref. 1
B-DECONCO	Contaminated concrete	Table B-27, Ref. 1
B-DETRASH	Combustible/compactible trash	Table A-45, this report
B-DERBSIN	Chelated ion exchange resins	Table A-44, this report
B-DEEVAPB	Evaporator bottoms	Table A-44, this report

PWR Waste Streams

Waste Stream	Description	Scaling References
P-DECORES	Activated core shroud	Table 5.1, Ref. 82
P-DEACINT	Activated reactor internals	Table 5.1, Ref. 82
P-DEACVES	Activated reactor vessels	Table 5.3, Ref. 82
P-DEACTCO	Activated concrete	Table 5.8 Ref. 82
P-DECONME	Contaminated metal	Table 5-27, Ref. 1
PUDECONO	Contaminated concrete	Table B-27, Ref. 1
P-DETRASH	Combustible/compactible trash	Table A-44, this report
P-DER[SIN	Chelated ion exchange resins	Table A-44, this report
P-DEFILCR	Filter cartridges	Table A-44, this report
P-DEEVAPB	Evaporator bottoms	Table A-44, this report

Radionuclide distributions for activated materials were obtained from reference 82, which presents a series of calculations on the buildup of neutron activation products within reactor materials such as internals, vessel walls, the concrete biological shield, and rebar. All calculations were carried out for the core axial midplane for reactors operating at 30 effective full power years. Illustrations of the assumed flux distributions within the reactors are shown in Figure A.10 for a PWR and Figure A.11 for a BWR (Ref. 82).

Distributions for the PWR and BWR core shroud (DECORES) were obtained by scaling to the concentration distributions for core shrouds as given in reference 82. A metal density of 7.8 g/cm^3 was assumed. Similarly, distributions for activated reactor vessels (DECAVES) were obtained By scaling to the vessel wall distriButions given in this reference. Activated reactor internals required more of an approximation. For PWRs a distribution corresponding to the core barrel in reference 82 was used. (A number of isotopic relationships were checked for the PWR core barrel, thermal pads, and vessel cladding, and were determined to be similar.) For BWRs, the shroud distribution was used.

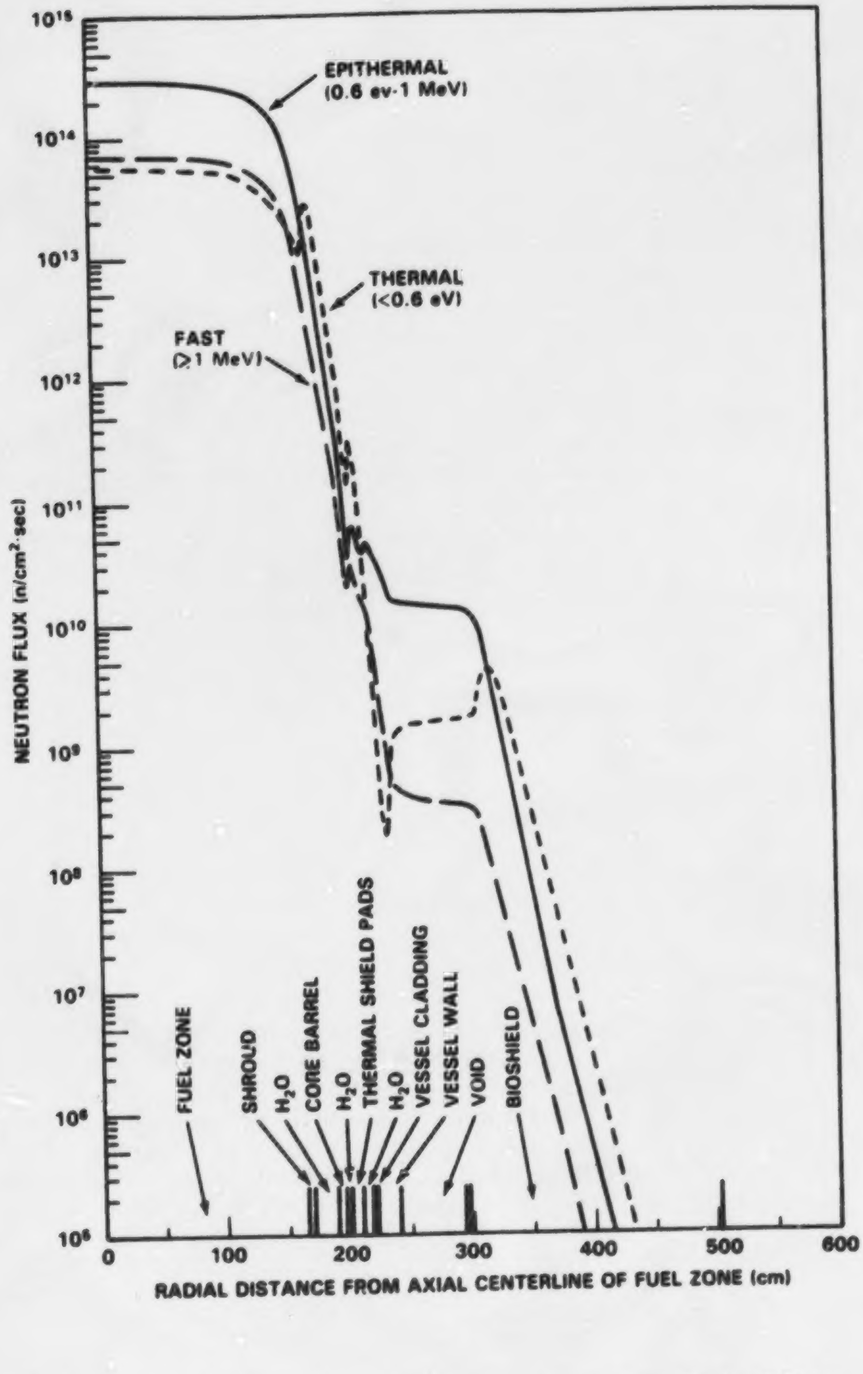

Figure A.10. Radial Three Group Neutron Flux Distribution at Core Axial Midplane for Westinghouse PWR

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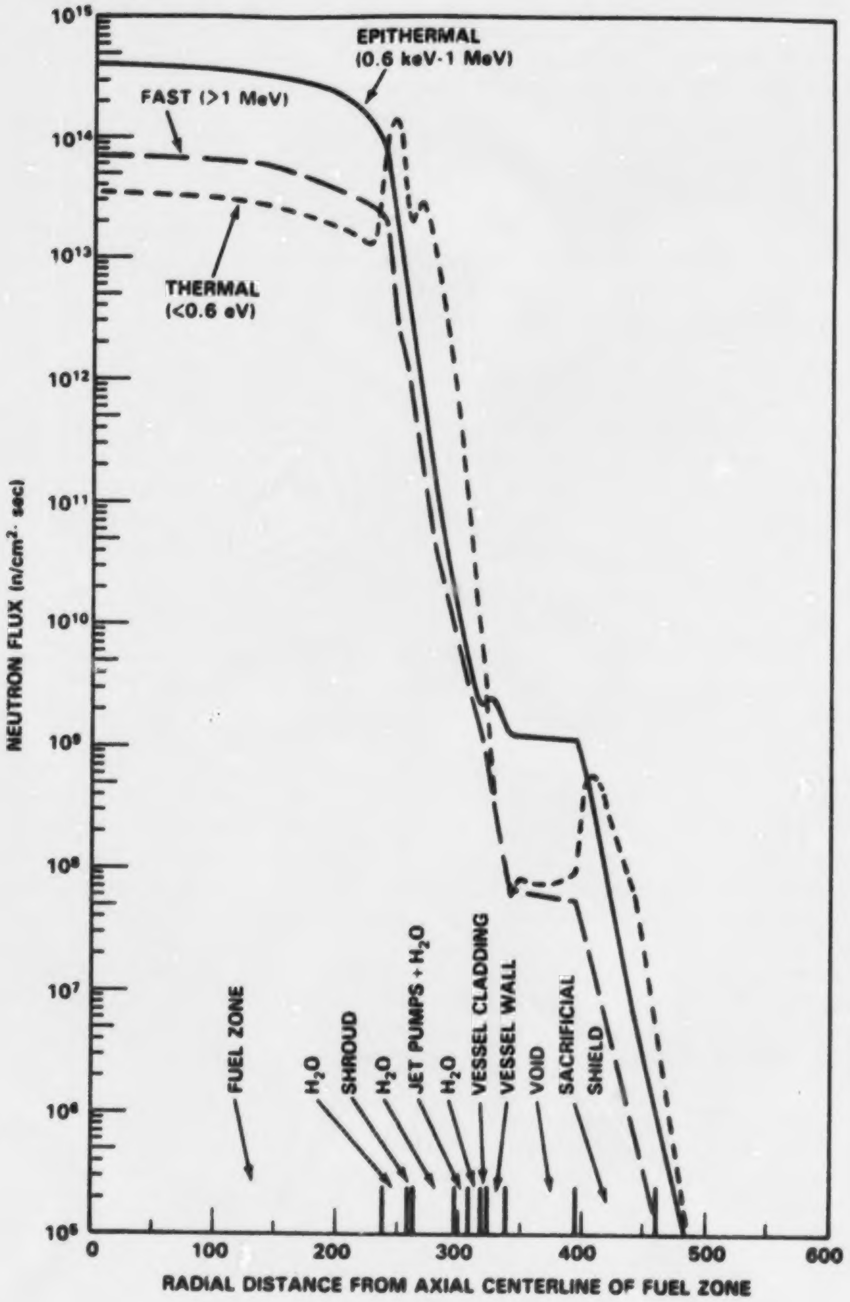

Figure A.11. Radial Three Group Neutron Flux Distribution at Core Axial Midplane for General Electric BWR

Activated concrete (DEACTCO) was assumed to consist of both high density concrete and rebar. For the radionuclide distributions in this report, the concrete and rebar distributions in reference 82 were used assuming an activation distance of 10 cm from the edge of the biological shield. Ten percent of the concrete mass was assumed to consist of rebar. The density of the concrete mass was assumed to be about 4.5 g/cm^3, which is based upon an assumed density of high density cement in the range of 250-300 lb/ft^3 and an assumed density of the rebar of about 7.8 g/cm^3.

Radionuclide distributions for contaminated metal and concrete were scaled from assumed distributions for reactor crud as obtained from reference 1. Distributions for the remaining waste streams were obtained by scaling from LWR waste streams generated during normal operations.

Sixteen individual radionuclide concentrations for all 19 LWR decommissioning waste streams are thus presented in Tables A-62 and A-63. Radionuclide concentrations for three waste streams (B-DEACINT, P-DEACINT, and P-DEACVES) are actually given as weighted averages across concentration distributions, while the concentrations for the remaining 16 waste streams are given as averages. The concentrations for very short lived radionuclides as listed in reference 82 have generally been deleted.

A.9.7.3 West Valley Demonstration Project

Individual radionuclide concentrations for the 25 waste streams identified as being generated during the West Valley Demonstration Project are estimated based upon information contained in Tables A-39 through A-42 in Section A.4.7.2, and also upon Tables A-64 through A-68 (Refs. 75 and 76). The radionuclide concentrations are given in an as-generated form prior to further processing steps such as compaction or solidification. Concentrations for 15 of the 25 waste streams are given as averages, and these are listed in Table A-69 in units of Ci/m^3. This table includes approximate unsolidified concentrations for the two high-level waste streams: W-THORHLW and W-PUREHLW. The data for the W-THORHLW stream only covers a few radionuclides, and additional research is needed to better characterize the range of radionuclides (particularly transuranic radionuclides) contained in this waste. The radionuclide concentrations for the W-PUREHLW waste stream are obtained by dividing the total estimate radionuclide activity (sludge plus supernate) by the approximate volume (2 million liters). Considerable uncertainties exist for several waste streams.

The remaining 10 waste streams (W-FRSRESN, W-RTSRESN, W-LTTRASH, W-HTTRASH, W-LTEQUIP, W-HTEQUIP, W-DDRACKS, W-DDLTRUB, W-DDHTRUB, and W-DDHTLQD) are given as concentration distributions. These concentration distributions are listed in Table A-70, and in each concentration range the percentage of the waste stream volume that falls into the given concentration range is listed. Also listed in parentheses for each concentration range is the average gross radionuclide concentration. The weighted average gross concentration across the entire distribution is also listed for each waste stream.

The radionuclide spectral distribution for each distributed waste stream is listed in Table A-71 in units of Ci/m^3. The spectral distributions are weighted to correspond to the weighted average concentrations listed in Table A-70. For most waste streams, the radionuclide spectral distributions were taken as appropriate multiples of Table A-65. For the W-HTTRASH, W-HTEQUIP, W-DDHTRUB, and W-DDHTLQD waste streams, however, a separate transuranic distribution is imposed.

Table A-62. Radionuclide Concentrations for PWR Decommissioning Waste Streams (Ci/m^3)

Radionuclide	P-DECORES	P-DEACINT	P-DEACVES	P-DEACTCO	P-DECONME	P-DECONCO	P-DETRASH	P-DERESIN	P-DEFILCR	P-DEEVAPB
H-3	8.66E+0	1.35E+0	1.16E-2	8.56E-1	1.10E-5	6.49E-7	7.05E-3	1.11E+2	3.46E-1	2.31E-1
C-14	2.17E+1	4.23E-1	6.53E-3	3.09E-4	5.26E-7	3.09E-8	2.61E-4	4.06E+0	1.28E-2	1.46E-2
Cl-36	4.42E-1	9.03E-3	1.43E-4	3.32E-10						
Ar-39	<1.30E-1	<1.13E-3	1.70E-5	1.24E-3						
Ca-41	4.08E-3	7.90E-5	1.25E-6	1.96E-3						
Mn-53	2.78E-3	2.96E-5	4.12E-7	2.93E-9						
Fe-55	1.82E+5	3.38E+3	5.28E+1	1.86E+0	4.16E-2	2.44E-3	1.39E-1	9.75E+1	1.67E+2	2.85E+1
Ni-59	9.53E+1	2.68E+0	3.84E-2	1.12E-5	4.61E-5	2.71E-6	1.65E-4	1.16E-1	1.99E-1	2.94E-2
Co-60	1.12E+5	1.97E+3	2.87E+1	7.95E-2	7.47E-2	4.39E-3	2.68E-1	1.89E+2	3.22E+2	4.76E+1
Ni-63	1.55E+4	3.35E+2	5.10E+0	1.41E-3	3.78E-3	2.22E-4	5.10E-2	3.59E+1	6.14E+1	6.45E-1
Se-79	8.29E-4	6.49E-6	8.95E-8	3.52E-10						
Sr-90	1.73E+0	<7.05E-4	<1.34E-4	<7.80E-6	4.38E-5	2.58E-6	5.15E-4	8.08E+0	2.53E-2	4.42E-2
Nb-92m	1.04E-6	9.17E-9	4.03E-11	<7.88E-14						
Zr-93	9.53E-5	5.50E-7	8.95E-9	3.05E-9						
Mo-93	8.14E-1	5.50E-3	6.89E-5	1.95E-7						
Nb-94	3.47E-1	4.09E-3	5.64E-5	<1.11E-6	1.46E-6	8.56E-8	5.51E-6	3.68E-3	6.29E-3	9.29E-4
Tc-99	1.12E-1	1.16E-3	1.34E-5	<4.42E-8	1.23E-8	7.21E-10	2.19E-6	3.43E-2	1.08E-4	9.38E-4
Ag-108m	<8.66E-2	<1.27E-3	<1.79E-5	<1.11E-6						
I-129	<5.20E-7	<1.97E-9	<3.76E-11	<2.36E-12	3.42E-8	2.10E-9	6.49E-6	1.02E-1	3.19E-4	2.49E-3
Cs-135	<3.47E-5	<1.27E-7	<2.42E-9	<1.50E-10	1.23E-8	7.21E-10	2.19E-6	3.43E-2	1.08E-4	9.38E-4
Cs-137	<1.73E+0	<7.05E-3	<1.34E-4	<8.13E-6	3.84E-4	2.26E-5	5.82E-2	9.13E+2	2.87E+0	2.49E+1
Sm-146	8.66E-11	1.83E-12	2.59E-14	<1.75E-14						
Sm-151	3.99E-3	6.35E-4	3.40E-6	<1.45E-4						
Tb-158	1.64E-3	2.40E-5	3.40E-7	<1.25E-8						
Ho-166m	1.39E-1	1.83E-8	2.50E-5	<4.21E-6						
Hf-178m	3.72E-2	3.93E-3	<3.31E-5	<1.93E-2						
Pb-205	1.55E-6	1.83E-8	2.59E-10	<7.37E-12						
U-233	<3.12E-4	<1.41E-5	<1.43E-7	<1.24E-6						
U-235					7.02E-8	4.13E-9	1.84E-7	1.96E-3	1.10E-4	1.29E-5
U-238					5.54E-7	3.25E-8	1.45E-6	1.55E-2	8.64E-4	1.02E-4
Np-237					1.35E-11	7.93E-13	3.54E-11	3.78E-7	2.11E-8	2.48E-9
Pu-238					1.37E-3	8.07E-5	1.39E-4	1.08E+0	7.56E-2	7.46E-2
Pu-239/40	<6.06E-3	<3.24E-4	<2.42E-6	<1.81E-6	1.81E-3	1.06E-4	1.29E-4	7.58E-1	1.14E-1	3.54E-2
Pu-241					3.36E-2	2.13E-3	5.61E-3	3.31E+1	5.00E+0	1.73E+0
Pu-242					3.97E-6	2.33E-7	2.82E-7	1.66E-3	2.51E-4	7.36E-5
Am-241					5.42E-6	3.19E-7	9.22E-5	7.79E-1	4.94E-2	4.50E-2
Am-243					3.68E-7	2.16E-8	6.23E-6	5.25E-2	3.31E-3	3.03E-3
Cm-243					3.55E-7	2.09E-8	6.38E-8	4.13E-4	5.81E-5	9.71E-5
Cm-244					3.36E-6	1.97E-7	6.08E-5	3.75E-1	3.31E-2	7.69E-2
Total:	3.10E+5	5.69E+3	8.67E+1	2.82E+0	1.57E-1	9.40E-3	5.30E-1	1.40E+3	5.59E+2	1.04E+2

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Table A-63. Radionuclide Concentrations for BWR Decommissioning Waste Streams (Ci/m^3)

Radionuclide	B-DECORES	B-DEACINT	B-DEACVES	B-DEACTCO	B-DECONME	B-DECONCO	B-DETRASH	B-DERESIN	B-DEEVAPB
H-3	1.73E+1	4.03E-1	9.80E-2	5.86E-1	3.93E-5	3.04E-6	1.52E-3	2.24E-2	1.69E-1
C-14	9.71E+0	2.26E-1	1.91E-2	2.12E-4	1.87E-6	1.45E-7	9.40E-5	1.39E-3	1.05E-2
Cl-36	2.13E-1	4.96E-3	3.91E-4	1.09E-5					
Ar-39	2.53E-2	5.88E-4	2.07E-4	1.30E-3					
Ca-41	1.86E-3	4.34E-5	3.56E-6	1.36E-3					
Mn-53	6.12E-4	1.42E-3	5.46E-6	4.05E-9					
Fe-55	7.85E+4	1.83E+3	1.60E+2	1.26E+0	1.48E-1	1.14E-2	1.35E-1	1.10E+0	2.05E+1
Ni-59	5.72E+1	1.33E+0	1.25E-1	7.66E-6	1.64E-4	1.27E-5	1.40E-4	1.14E-3	2.12E-2
Co-60	4.26E+4	9.91E+2	9.39E+1	5.19E-2	2.66E-1	2.05E-2	2.27E-1	1.86E+0	3.44E+1
Ni-63	7.58E+3	1.77E+2	1.56E+1	9.53E-4	1.35E-2	1.04E-3	3.06E-3	2.51E-2	4.66E-1
Se-79	1.33E-4	3.10E-6	6.69E-7	2.74E-10					
Sr-90	<2.00E-1	<4.65E-3	<9.80E-5	<5.23E-6	1.56E-4	1.21E-5	2.86E-4	4.25E-3	3.20E-4
Nb-92m	5.99E-8	1.39E-9	1.48E-9	<2.34E-13					
Zr-93	1.33E-5	3.10E-7	4.69E-8	<2.01E-9					
Mo-93	1.02E-1	2.38E-3	3.01E-4	1.03E-7					
Nb-94	8.40E-2	1.95E-3	1.91E-4	<6.53E-7	5.19E-6	4.01E-7	4.42E-6	3.61E-5	6.71E-4
Tc-99	2.00E-2	4.65E-4	5.86E-5	2.34E-8	4.37E-8	3.37E-9	6.03E-6	8.91E-5	6.77E-4
Ag-108m	<2.66E-2	<6.19E-4	<5.86E-5	<7.03E-7					
I-129	<5.59E-8	<1.30E-9	<2.97E-11	<1.63E-12	1.22E-7	9.41E-9	1.61E-5	2.38E-4	1.80E-3
Cs-135	<3.59E-6	<8.36E-8	<2.50E-9	<1.06E-10	4.37E-8	3.37E-9	6.03E-6	8.91E-5	6.77E-4
Cs-137	<2.00E-1	<4.65E-3	<1.37E-4	<5.66E-6	1.37E-3	1.06E-4	1.61E-1	2.38E+0	1.80E+1
Sm-146	3.86E-11	8.98E-13	3.09E-13	<2.25E-14					
Sm-151	5.06E-3	1.18E-4	1.21E-4	<8.88E-5					
Tb-158	5.06E-4	1.18E-5	4.30E-6	<1.73E-8					
Ho-166m	3.73E-2	8.67E-4	7.40E-5	<2.36E-6					
Hf-178m	4.92E-2	<1.15E-3	<2.93E-4	<1.06E-5					
Pb-205	3.86E-7	8.98E-9	1.76E-9	<2.38E-11					
U-233	<2.13E-4	<4.96E-6	<8.63E-7	<1.04E-7					
U-235					2.49E-7	1.93E-8	2.75E-8	6.21E-8	9.28E-6
U-238					1.97E-6	1.52E-7	2.16E-7	4.91E-7	7.34E-5
Np-237					4.80E-11	3.72E-12	5.29E-12	1.19E-11	1.79E-9
Pu-238					4.89E-3	3.77E-4	5.16E-5	9.72E-5	5.38E-2
Pu-239/40	<3.59E-3	<8.36E-5	<2.27E-5	<9.19E-7	6.44E-3	4.97E-4	2.61E-5	6.21E-5	2.55E-2
Pu-241					1.29E-1	9.98E-3	1.27E-3	3.04E-3	1.24E+0
Pu-242					1.41E-5	1.09E-6	5.68E-8	1.37E-7	5.58E-5
Am-241					1.93E-5	1.49E-6	2.18E-5	2.71E-5	3.25E-2
Am-243					1.31E-6	1.01E-7	1.47E-6	1.83E-6	2.20E-3
Cm-243					1.26E-6	9.77E-8	4.35E-8	3.15E-8	7.01E-5
Cm-244					1.20E-5	9.24E-7	3.39E-4	2.12E-5	5.55E-2
Total:	1.29E+5	3.00E+3	2.70E+2	1.90E+0	5.70E-1	4.39E-2	5.30E-1	5.40E+0	7.50E+1

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Table A-64. Summary of West Valley Demonstration Project As-Generated Waste Streams, Concentrations, and Volumes

Waste Description	Symbol	As-Generated Concentration (Ci/m^3)**
Thorex HLW*	W-THORHLW	Table A-67 ÷ 47 m^3
Purex HLW	W-PUREHLW	Table A-68 ÷ 2,000 m^3
Trash from existing systems	W-COTRASH	3.2E-4 x Table A-65
Miscellaneous dry solids	W-NCSOLID	1.4E-3 x Table A-65
LLWTF sludge and resins	W-LLWTFRE	3.8%: 6.02E-3 x Table A-65 96.2%: 7.42E-3 x Table A-65
FRS filter precoat & resins	W-FRSRESN	13.3%: 2.52 Cs-137 & .28 Sr-90 86.7%: 22.4 Cs-137 & 2.1 Sr-90
RTS liquid waste	W-FRSLIQD	3.36 Cs-137 & 0.38 Sr-90
RTS filter backwash & resins	W-RTSRESN	19.4%: .66 Cs-137 & .07 Sr-90 80.6%: 3.5 Cs-137 & .39 Sr-90
Trash, low TRU content	W-LTTRASH	77%: .0066 x Table A-65 23%: .04 x Table A-65
Trash, high TRU content	W-HTTRASH	70.5%: .0066 x Table A-65 # 29.5%: .066 x Table A-65 ##
Equip. & hardware, low TRU	W-LTEQUIP	90.9%: 2.5 x Table A-65 9.1%: 36 x Table A-65
Equip. & hardware, high TRU	W-HTEQUIP	22.1%: 2.5 x Table A-65 # 77.9%: 25 x Table A-65 ##
PD liquid waste	W-PDWLIQD	+
Vit. supernate	W-VITSUPR	Table A-66
Vit. sludge wash	W-VITWASH	Table A-66
Vit. scrub condensate	W-VITSCRB	70 Cs-137, 0.7 Sr-90
Vit. melter feed overheads	W-VITMELT	1,540 Cs-137, 1231 Sr-90
Vit. frac. condensate	W-VITFRAC	++
Vit. zeolite slurry	W-VITZEOL	+++
D/D fuel storage racks	W-DDRACKS	90.9%: 0.05 x Table A-65 9.1%: 0.25 x Table A-65
D/D rubble, low TRU	W-DDLTRUB	90.9%: 5E-3 x Table A-65 9.1%: 0.05 x Table A-65
D/D rubble, high TRU	W-DDHTRUB	88.5%: 5E-3 x Table A-65 # 11.5%: 5E-2 x Table A-65 ##
D/D liquid, low TRU	W-DDLTLQD	90.9%: 60.2 x Table A-65 9.1%: 168. x Table A-65
D/D liquid, high TRU	W-DDHTLQD	20.3%: 60.2 x Table A-65 # 79.7%: 602 x Table A-65 ##
D/D resins	W-DDRESIN	196 x Table A-65

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Table A-64. Summary of West Valley Demonstration Project As-Generated Waste Streams, Concentrations, and Volumes (continued)

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*Abbreviations--HLW: high-level waste, LLWTF: low-level waste treatment system, FRS: fuel receiving and storage, RTS: radwaste treatment system, TRU: transuranic, equip.: equipment, PD: presolidification decontamination, Vit.: vitrification, frac.: fractionator, D/D: decontamination and decommissioning.

**As-generated concentrations for trash waste streams were obtained by dividing data in reference 76 by 3; as-generated concentrations for liquids and other wet wastes were obtained by multiplying data in reference 76 by 1.4.

***As-generated volumes for trash waste streams were obtained by multiplying data in reference 76 by 3; as-generated volumes for liquids and other wet wastes were obtained by dividing data in reference 76 by 1.4.

#Except TRU radionuclides. As-packaged TRU radionuclides are 55 nCi/gm.

##Except TRU radionuclides. As-packaged TRU radionuclides are 550 nCi/gm.

+As-solidified total: 36 due to Pu-238 to 241, Am-241, U-235 and 238, Sr-90, Cs-137, Tc-99.

++As-solidified concentration: Cs-137: 4.4E-2, remainder 4.8E-4 (Sr-90, Cs-134 and 135, Pu-238 ...)

+++As-solidified concentration: Cs-137: 2.7, remainder: 2.9E-3 (Sr-90, Cs-134 and 135, Pu-238 ...)

Table A-65. Reference Radionuclide Distribution for Most West Valley Demonstration Project Waste Streams (Ci/m^3)

Radionuclide	Activity	Radionuclide	Activity
H-3	1.5E-8	Ba-137m	9.4E-1
C-14	1.7E-8	Pm-147	1.4E-1
Ni-63	1.2E-7	Sm-151	2.5E-2
Se-79	4.7E-6	Eu-152	6.6E-5
Sr-90	9.6E-1	Eu-154	2.5E-5
Y-90	9.6E-1	Eu-155	5.6E-3
Zr-93	2.E-5	Np-237	1.4E-6
Nb-93m	2.8E-5	Np-239	2.9E-5
Tc-99	1.0E-4	Pu-238	1.1E-3
Ru-106	4.4E-4	Pu-239	2.0E-4
Rh-107	4.4E-4	Pu-240	1.5E-4
Pd-107	6.4E-8	Pu-241	1.4E-2
Cd-113	4.4E-17	Pu-242	2.1E-7
Sb-235	1.9E-3	Am-241	1.5E-3
Te-125m	4.2E-4	Am-242	2.6E-5
Sn-126	5.0E-6	Am-243	2.9E-5
Sb-126m	3.4E-7	Cm-242	2.6E-5
Sb-126	3.4E-7	Cm-243	4.2E-6
I-129	2.6E-11	Cm-244	1.3E-3
Cs-134	8.3E-4	Cm-245	2.1E-7
Cs-135	1.4E-5	Cm-246	2.5E-8
Cs-137	1.0E+0	Total:	4.078

Table A-66. Radionuclide Concentrations of As-Generated W-VITSUPR and W-VITWASH Waste Streams

	As-Generated Concentration (Ci/m^3)	
Radionuclide	W-VITSUPR	W-VITWASH
H-3	1.4E-4	
C-14	1.4E-4	
Ni-63	5.6E-4	
Se-79	2.5E-2	6.9E-1
Sr-90	3.6E-2	2.0E-1
Y-90	3.6E-2	
Zr-93	1.4E-4	
Nb-93m	1.4E-4	
Tc-99	1.1E-1	2.9E+0
Ru-106	2.4E-2	1.3E+0
Rh-106	2.4E-2	
Sb-125	1.1E-3	
Te-125m	2.8E-4	
Sn-126	2.8E-4	
Cs-134	4.2E-4	
Cs-137	5.3E-1	2.9E+1
Ba-137m	5.0E-1	
Pm-147	2.0E-1	5.2E+0
Sm-151	5.3E-3	1.4E-1
Eu-152	3.8E-2	1.0E+0
Eu-154	6.0E-2	1.7E+0
Eu-155	1.1E-1	2.9E+0
Pu-238	9.0E-3	2.4E-1
Pu-239	1.7E-3	
Pu-240	1.3E-3	
Pu-241	1.3E-1	1.8E-1

Table A-67. Estimated 1987 Radioactivity of Acidic Thorex Wastes (Tank 8D4)

Fission Product	Radioactivity (curies)
Cobalt-60	1.5×10^3
Strontium-90	6.5×10^5
Yttrium-90	6.5×10^5
Cesium-134	5.4×10^2
Cesium-137	6.9×10^5
Barium-137m	6.4×10^5
Europium-154	4.2×10^3
Total	2.6×10^6

Source: Ref. 75.

Table A-68. Estimated 1987 Radioactivity of Neutralized Purex Wastes (Tank 8D2)

Fission Product	Radioactivity (curies) Sludge	Supernatant	Total
Tritium (H-3)	0	2.4×10^3	2.4×10^3
Selenium-79	0	5.0×10^1	5.0×10^1
Strontium-90	6.6×10^6	6.7×10^4	6.7×10^6
Yttrium-90	6.6×10^6	6.7×10^4	6.7×10^6
Zirconium-93	2.5×10^2	<1	2.5×10^2
Niobium-93m	2.4×10^2	<1	2.4×10^2
Technetium-99	0	1.9×10^3	1.9×10^3
Ruthenium-106	1.0×10^2	1.0×10^1	1.1×10^2
Rhodium-106	1.0×10^2	1.0×10^1	1.1×10^2
Palladium-107	6	<1	6
Antimony-125	6.0×10^3	6.0×10^1	6.1×10^3
Tellurium-125m	0	6.1×10^3	6.1×10^3
Tin-126	4.0×10^1	<1	4.0×10^1
Antimony-126m	4.0×10^1	<1	4.0×10^1
Antimony-126	4.0×10^1	<1	4.0×10^1
Iodine-129†	0	4.7	4.7
Cesium-134	0	2.1×10^4	2.1×10^4
Cesium-135	0	3.5×10^1	3.5×10^1
Cesium-137	0	8.9×10^6	8.9×10^6
Barium-137m	0	8.4×10^6	8.4×10^6
Cerium-144	1.1×10^1	<1	1.1×10^1
Praseodymium-144	1.1×10^1	<1	1.1×10^1
Promethium-147	6.0×10^4	6.0×10^1	6.0×10^4
Samarium-151	2.0×10^5	2.0×10^1	2.0×10^5
Europium-152	4.1×10^2	<1	4.1×10^2
Europium-154	1.3×10^5	1.4×10^3	1.3×10^5
Total			3.1×10^7

Actinide	Radioactivity (curies) Sludge	Supernatant	Total
Uranium-235	8.0×10^{-2}	0	8.0×10^{-2}
Uranium-238	8.2×10^{-1}	0	8.2×10^{-1}
Neptunium-237	2.3×10^1	<1	2.3×10^1
Neptunium-239	2.2×10^2	<1	2.2×10^2
Plutonium-238	1.5×10^3	1.5	1.5×10^3
Plutonium-239	1.8×10^3	1.8	1.8×10^3
Plutonium-240	9.7×10^2	<1	9.7×10^2
Plutonium-241	7.0×10^4	7.0×10^1	7.0×10^4
Plutonium-242	<1	0	<1
Americium-241	2.0×10^4	2.0×10^1	2.0×10^4
Americium-242	1.8×10^2	<1	1.8×10^2
Americium-242m	1.8×10^2	<1	1.8×10^2
Americium-243	2.2×10^2	<1	2.2×10^2
Curium-242	<1	0	<1
Curium-244	8.8×10^3	8.8	8.8×10^3
Curium-245	1	0	1
Curium-246	<1	0	<1
Total			1.0×10^5

† No iodine should remain after Purex reprocessing of the fuel; however, the iodine was included to yield conservative doses.

Source: Ref. 75.

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Table A-69. As-Generated Radionuclide Concentrations (Ci/m³) for Fifteen West Valley Demonstration Project Waste Streams

Radionuclide	W-THORHLW	W-PUREHLW	W-COTRASH	W-NCSOLID	W-FRSLIQD	W-PDWLIQD	W-VITSUPR	W-VITWASH
H-3		1.20E+0	4.80E-12	2.10E-11			1.40E-4	
C-14			5.44E-12	2.38E-11			1.40E-4	
Co-60	3.19E+1							
Ni-63			3.84E-11	1.68E-10			5.60E-4	
Se-79		2.50E-2	1.50E-9	6.58E-9			2.50E-2	6.90E-1
Sr-90	1.38E+4	3.35E+3	3.07E-4	1.34E-3	3.80E-1	2.45E+1	3.60E-2	2.00E-1
Y-90	1.38E+4	3.35E+3	3.07E-4	1.34E-3			3.60E-2	
Zr-93		1.25E-1	8.96E-9	3.98E-8			1.40E-4	
Nb-93m		1.20E-1	8.96E-9	3.98E-8			1.40E-4	
Tc-99		9.50E-1	3.20E-8	1.40E-7		2.55E-3	1.10E-1	2.90E+0
Ru-106		5.50E-2	1.41E-7	6.16E-7			2.40E-2	1.30E+0
Rh-106		5.50E-2	1.41E-7	6.16E-7			2.40E-2	
Pd-107		3.00E-3	2.05E-11	8.96E-11				
Cd-113			1.41E-20	6.16E-20				
Sb-125		3.05E+0	6.08E-7	2.66E-6			1.10E-3	
Te-125m		3.05E+0	1.34E-7	5.88E-7			2.80E-4	
Sn-126		2.00E-2	1.60E-9	7.00E-9			2.80E-4	
Sb-126m		2.00E-2	1.09E-10	4.76E-10				
Sb-126		2.00E-2	1.09E-10	4.76E-10				
I-129		2.35E-3	8.32E-15	3.64E-14				
Cs-134	1.15E+1	1.05E+1	2.66E-7	1.16E-6			4.20E-4	
Cs-135		1.75E-2	4.48E-9	1.96E-8				
Cs-137	1.47E+4	4.45E+3	3.20E-4	1.40E-3	3.36E+0	2.55E+1	5.30E-1	2.90E+1
Ba-137m	1.36E+4	4.20E+3	3.01E-4	1.32E-3			5.00E-1	
Ce-144		5.50E-3						
Pr-144		5.50E-3						
Pm-147		3.00E+1	4.48E-5	1.96E-4			2.00E-1	5.20E+0
Sm-151		1.00E+2	8.00E-6	3.50E-5			5.30E-3	1.40E-1
Eu-152		2.05E-1	2.11E-8	9.24E-8			3.80E-2	1.00E+0
Eu-154	8.94E+1	6.50E+1	8.00E-6	3.50E-5			6.00E-2	1.70E+0
Eu-155			1.79E-6	7.84E-6			1.10E-1	2.90E+0

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Table A-69. (continued)

Radionuclide	W-THORHLW	W-PUREHLW	W-COTRASH	W-NCSOLID	W-FRSLIQD	W-PDWLIQD	W-VITSUPR	W-VITWASH
U-235	4.00E-5							
U-238	4.10E-4							
Np-237	1.15E-2	4.48E-10	1.96E-9					
Np-239	1.10E-1	9.28E-9	4.06E-8					
Pu-238		7.50E-1	3.52E-7	1.54E-6		2.80E-2	9.00E-3	2.40E-1
Pu-239/40		1.39E+0	1.12E-7	4.90E-7		8.92E-3	3.00E-3	
Pu-241		3.50E+1	4.48E-6	1.96E-5		3.57E-1	1.30E-1	1.80E-1
Pu-242		5.00E-4	6.72E-11	2.94E-10				
Am-241		1.00E+1	4.80E-7	2.10E-6		3.82E-2		
Am-242		9.00E-2	8.32E-9	3.64E-8				
Am-242m		9.00E-2						
Am-243		1.10E-1	9.28E-9	4.06E-8				
Cm-242		5.00E-4	8.32E-9	3.64E-8				
Cm-243			1.34E-9	5.88E-9				
Cm-244		4.40E+0	4.16E-7	1.82E-6				
Cm-245		5.00E-4	6.72E-11	2.94E-10				
Cm-246		5.00E-4	8.00E-12	3.50E-11				
Total:	5.60E+4	1.56E+4	1.30E-3	5.70E-3	3.74E+0	5.04E+1	1.84E+0	4.55E+1

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Table A-69. (continued)

Radionuclide	W-VITSCRB	W-VITMELT	W-VITFRAC	W-VITZEOL	W-DDRESIN	W-LLWTRFE	W-DDLTLQD
H-3					2.94E-6	1.11E-10	1.05E-6
C-14					3.33E-6	1.25E-10	1.19E-6
Co-60							
Ni-63					2.35E-5	8.84E-10	8.40E-6
Se-79					9.21E-4	3.46E-8	3.29E-4
Sr-90	7.00E-1	1.23E+3	6.59E-4	3.99E-3	1.88E+2	7.07E-3	6.72E+1
Y-90					1.88E+2	7.07E-3	6.72E+1
Zr-93					5.49E-3	2.06E-7	1.96E-3
Nb-93m					5.49E-3	2.06E-7	1.96E-3
Tc-99					1.96E-2	7.37E-7	7.00E-3
Ru-106					8.62E-2	3.24E-6	3.08E-2
Rh-106					8.62E-2	3.24E-6	3.08E-2
Pd-107					1.25E-5	4.71E-10	4.48E-6
Cd-113					8.62E-15	3.24E-19	3.08E-15
Sb-125					3.72E-1	1.40E-5	1.33E-1
Te-125m					8.23E-2	7.37E-3	2.94E-2
Sn-126					9.80E-4	3.68E-8	3.50E-4
Sb-126m					6.66E-5	2.50E-9	2.38E-5
Sb-126					6.66E-5	2.50E-9	2.38E-5
I-129					5.10E-9	1.92E-13	1.82E-9
Cs-134			5.70E-7	3.44E-6	1.63E-1	6.11E-6	5.81E-2
Cs-135			9.62E-9	5.81E-8	2.74E-3	1.03E-7	9.80E-4
Cs-137	7.00E+1	1.54E+3	6.16E-2	3.78E+0	1.96E+2	7.37E-3	7.00E+1
Ba-137m					1.84E+2	6.92E-3	6.58E+1
Ce-144							
Pr-144							
Pm-147					2.74E+1	1.03E-3	9.80E+0
Sm-151					4.90E+0	1.84E-4	1.75E+0
Eu-152					1.29E-2	4.86E-7	4.62E-3
Eu-154					4.90E+0	1.84E-4	1.75E+0
Eu-155					1.10E+0	4.13E-5	3.92E-1
U-235							
U-238							

Table A-69. (continued)

Radionuclide	W-VITSCRB	W-VITMELT	W-VITFRAC	W-VITZEOL	W-DDRESIN	W-LLWTRFE	W-DDLTLQD
Np-237					2.74E-4	1.03E-8	9.80E-5
Np-239					5.68E-3	2.14E-7	2.03E-3
Pu-238			7.56E-7	4.56E-6	2.16E-1	8.10E-6	7.70E-2
Pu-239/40			2.40E-7	1.45E-6	6.86E-2	2.58E-6	2.45E-2
Pu-241			9.62E-6	5.81E-5	2.74E+0	1.03E-4	9.80E-1
Pu-242					4.12E-5	1.55E-9	1.47E-5
Am-241			1.03E-6	6.23E-6	2.94E-1	1.11E-5	1.05E-1
Am-242					5.10E-3	1.92E-7	1.82E-3
Am-242m							
Am-243					5.68E-3	2.14E-7	2.03E-3
Cm-242					5.10E-3	1.92E-7	1.82E-3
Cm-243					8.23E-4	3.09E-8	2.94E-4
Cm-244					2.55E-1	9.58E-6	9.10E-2
Cm-245					4.12E-5	1.55E-9	1.47E-5
Cm-246					4.90E-6	1.84E-10	1.75E-6
Total:	7.07E+1	2.77E+3	6.23E-2	3.78E+0	7.99E+2	3.74E-2	2.86E+2

Table A-70. Radionuclide Concentration Distributions (%) for Distributed West Valley Demonstration Project Waste Streams

Concentration Range (Ci/m³)	W-FRSRESN	W-RTSRESN	W-LTTRASH	W-HTTRASH	W-LTEQUIP	W-HTEQUIP	W-DDRACKS	W-DDLTRUB	W-DDHTRUB	W-DDHTLQD
0.01 - 0.1			77 (.0269)	70.5 (0.034)				90.9 (.0204)		
0.1 - 1.		19.4 (0.73)	23 (0.163)	29.5 (0.34)			90.9 (0.204)	9.1 (0.204)	88.5 (0.13)	
1. - 10.	13.3 (2.80)	80.6 (3.89)					9.1 (1.02)		11.5 (1.30)	
10. - 100.	86.7 (24.5)				90.9 (10.2)	22.1 (10.3)				
100. - 1000.					9.1 (147.)	77.9 (103.)				20.3 (245.)
1000. - 10,000.										79.7 (2,453)
Weighted Average (Ci/m³)	2.16E+1	3.28	5.82E-2	1.24E-1	2.26E+1	8.25E+1	2.78E-1	3.71E-2	2.65E-1	2.00E+3

Table A-71. As-Generated Radionuclide Concentrations (Ci/m^3) for Distributed West Valley Demonstration Project Waste Streams

Radionuclide	W-FRSRESN	W-RTSRESN	W-LTTRASH	W-HTTRASH	W-LTEQUIP	W-HTEQUIP	W-DDRACKS	W-DDLTRUB	W-DDHTRUB	W-DDHTLQD
H-3			2.14E-10	3.62E-10	8.32E-8	3.00E-7	1.02E-9	1.36E-10	1.53E-10	7.38E-6
C-14			2.43E-10	4.10E-10	9.43E-8	3.41E-7	1.16E-9	1.55E-10	1.73E-10	8.36E-6
Ni-63			1.71E-9	2.89E-9	6.66E-7	2.40E-6	8.18E-9	1.09E-9	1.22E-9	5.90E-5
Se-79			6.71E-8	1.13E-7	2.61E-5	9.41E-5	3.21E-7	4.27E-8	4.78E-8	2.31E-3
Sr-90	1.86E+0	3.28E-1	1.37E-2	2.32E-2	5.33E+0	1.92E+1	6.55E-2	8.73E-3	9.77E-3	4.72E+2
Y-90			1.37E-2	2.32E-2	5.33E+0	1.92E+1	6.55E-2	8.73E-3	9.77E-3	4.72E+2
Zr-93			4.00E-7	6.75E-7	1.55E-4	5.61E-4	1.91E-6	2.55E-7	2.85E-7	1.38E-2
Nb-93m			4.00E-7	6.75E-7	1.55E-4	5.61E-4	1.91E-6	2.55E-7	2.85E-7	1.38E-2
Tc-99			1.43E-6	2.41E-6	5.55E-4	2.00E-3	6.82E-6	9.10E-7	1.02E-6	4.92E-2
Ru-106			6.28E-6	1.06E-5	2.44E-3	8.81E-3	3.00E-5	4.00E-6	4.48E-6	2.16E-1
Rh-106			6.28E-6	1.06E-5	2.44E-3	8.81E-3	3.00E-5	4.00E-6	4.48E-6	2.16E-1
Pd-107			9.14E-10	1.54E-9	3.55E-7	1.28E-6	4.36E-9	5.82E-10	6.52E-10	3.15E-5
Cd-113			6.28E-19	1.06E-18	2.44E-16	8.81E-16	3.00E-18	4.00E-19	4.48E-19	2.16E-14
Sb-125			2.17E-5	4.58E-5	1.05E-2	3.81E-2	1.30E-4	1.73E-5	1.93E-5	9.35E-1
Te-125m			6.00E-6	1.01E-5	2.33E-3	9.41E-3	2.86E-5	3.82E-6	4.28E-6	2.07E-1
Sn-126			7.14E-8	1.21E-7	2.77E-5	1.00E-4	3.41E-7	4.55E-8	5.09E-8	2.46E-3
Sb-126m			4.86E-9	8.20E-9	1.89E-6	6.81E-6	2.32E-8	3.09E-9	3.46E-9	1.67E-4
Sb-126			4.86E-9	8.20E-9	1.89E-6	6.81E-6	2.32E-8	3.09E-9	3.46E-9	1.67E-4
I-129			3.71E-13	6.27E-13	1.44E-10	5.21E-10	1.77E-12	2.36E-13	2.65E-13	1.28E-8
Cs-134			1.19E-5	2.00E-5	4.61E-3	1.66E-2	5.66E-5	7.55E-6	8.45E-6	4.08E-1
Cs-135			2.00E-7	3.38E-7	7.77E-5	2.80E-4	9.55E-7	1.27E-7	1.43E-7	6.89E-3
Cs-137	1.98E+1	2.95E+0	1.43E-2	2.41E-2	5.55E+0	2.00E+1	6.82E-2	9.10E-3	1.02E-2	4.92E+2
Ba-137m			1.34E-2	2.27E-2	5.22E+0	1.88E+1	6.41E-2	8.55E-3	9.57E-3	4.62E+2
Pm-147			2.00E-3	3.38E-3	7.77E-1	2.80E+0	9.55E-3	1.27E-3	1.43E-3	6.89E+1
Sm-151			3.57E-4	6.03E-4	1.39E-1	5.01E-1	1.71E-3	2.27E-4	2.55E-4	1.23E+1
Eu-152			9.42E-7	1.59E-6	3.66E-4	1.32E-3	4.50E-6	6.00E-7	6.72E-7	3.25E-2
Eu-154			3.57E-4	6.03E-4	1.39E-1	5.01E-1	1.71E-3	2.27E-4	2.55E-4	1.23E+1
Eu-155			8.00E-5	1.35E-4	3.11E-2	1.12E-1	3.82E-4	5.09E-5	5.70E-5	2.76E+0
Np-237			2.00E-8	8.54E-6	7.77E-6	2.88E-4	9.55E-8	1.27E-8	7.32E-5	1.47E-4
Np-239			4.14E-7	6.99E-7	1.61E-4	5.81E-4	1.98E-6	2.64E-7	2.95E-7	1.43E-2
Pu-238			1.57E-5	6.71E-3	6.10E-3	2.26E-1	7.50E-5	1.00E-5	5.75E-2	1.15E-1
Pu-239/40			5.00E-6	2.13E-3	1.94E-3	7.20E-2	2.39E-5	3.18E-6	1.83E-2	3.67E-2

Table A-71. As-Generated Radionuclide Concentrations (Ci/m³) (continued)

Radionuclide	W-FRSRESN	W-RRTSRESN	W-LTTRASH	W-HTTRASH	W-LTEQUIP	W-HTEQUIP	W-DDRACKS	W-DDLTRUB	W-DDHTRUB	W-DDHTLQ
Pu-241			2.00E-4	3.38E-4	7.77E-2	2.80E-1	9.55E-4	1.27E-4	1.43E-4	6.89E+0
Pu-242			3.00E-9	1.28E-6	1.17E-6	4.32E-5	1.43E-8	1.91E-9	1.10E-5	2.20E-5
Am-241			2.14E-5	9.15E-3	8.32E-3	3.08E-1	1.02E-4	1.36E-5	7.84E-2	1.57E-1
Am-242			3.71E-7	6.27E-7	1.44E-4	5.21E-4	1.77E-6	2.36E-7	2.65E-7	1.28E-2
Am-243			4.14E-7	1.77E-4	1.61E-4	5.96E-3	1.98E-6	2.64E-7	1.52E-3	3.04E-3
Cm-242			3.71E-7	6.27E-7	1.44E-4	5.21E-4	1.77E-6	2.36E-7	2.65E-7	1.28E-2
Cm-243			6.00E-8	2.56E-5	2.33E-5	8.64E-4	2.86E-7	3.82E-8	2.19E-4	4.14E-4
Cm-244			1.86E-5	7.93E-3	7.21E-3	2.67E-1	8.87E-5	1.18E-5	6.79E-2	1.36E-1
Cm-245			3.00E-9	1.28E-6	1.17E-6	4.32E-5	1.43E-5	1.91E-9	1.10E-5	2.20E-5
Cm-246			3.57E-10	1.52E-7	1.39E-7	5.14E-6	1.71E-9	2.27E-10	1.31E-6	2.62E-6
Total:	2.17E+1	3.28E+0	5.82E-2	1.24E-1	2.26E+1	8.24E+1	2.78E-1	3.71E-2	2.65E-1	2.00E+3

each of these four waste streams, a portion of the as-packaged waste is assumed to have a transuranic content (alpha-emitting nuclides having half-lives exceeding 5 years) of 55 nCi/gm. The remaining as-packaged portion of each of these four waste streams is assumed to have a transuranic content of 550 nCi/gm.

To determine radionuclide spectral distributions, appropriate multiples of Table A-65 were assumed for all radionuclides other than transuranic alpha-emitting nuclides having half-lives greater than 5 years. For the latter radionuclides, an equivalent volume concentration (Ci/m^3) was obtained using unpackaged ("untreated") waste densities of 0.13 g/cm^3 for W-HTTRASH, 1 g/cm^3 for W-DDHTLQD, and 2 g/cm^3 for W-HTEQUIP and W-DDHTRUB (Ref. 76). The spectral distribution for the radionuclides was then obtained by multiplying this equivalent volume concentration by the activity fractions of the radionuclides as determined from Table A-65. These activity fractions are as follows:

Activity Fractions

Np-237	3.27E-4	Am-243	6.77E-3
Pu-238	2.57E-1	Cm-243	9.80E-4
Pu-239/40	8.17E-2	Cm-244	3.03E-1
Pu-242	4.90E-5	Cm-245	4.90E-5
Am-241	3.50E-1	Cm-246	5.83E-6

A.9.8 Other Waste

These wastes consist of spent nuclear reactor fuel. For illustration, this waste source is assumed to consist of two separate waste streams: removed hardware (L-FUEHARD) and separated spent fuel rods (L-SPENTFU). This is based on the assumption that spent fuel consolidation activities are carried out prior to disposal. Concentrations for the removed hardware are taken to be the same as those for the R-FUEHARD waste stream as given in Section A.9.7.1. These concentrations are based on an assumed delay time of 1.5 years after removal from the reactor and a mix of PWR (33,000 MWd/MTHM burnup) and BWR (27,000 MWd/MTHM) at a 2:1 ratio. Hardware components are 302-304 stainless steel, Inconel, and Nicrobraz-50. These concentrations are as follows (Ref. 59):

L-FUEHARD

Radionuclide	Concentration (Ci/m^3)
C-14	8.9E-1
Fe-55	7.1E+4
Co-60	7.1E+4
Ni-59	5.4E+1
Ni-63	7.4E+3
Sr-90	1.8E-2
Nb-94	1.8E-3
Tc-99	1.3E-1
Total	1.49E+5

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The radionuclide inventory for the remaining fuel rods is given in Table A-72 in units of Ci/MTHM. This inventory is based on the assumed PWR/BWR mix and burnup as discussed above and includes the activated zircaloy cladding. Conversion to a volume or mass concentration depends upon the degree to which the spent fuel rods are consolidated. Using Table A-43, the total mass per assembly, assuming removal of the hardware, is about 307.5 Kg per BWR assembly or 631.8 Kg per PWR assembly. Each reference BWR assembly contains 0.1833 MTHM, while each reference PWR assembly contains 0.4614 MTHM. The gross volume of the fuel rod array in the assembly (less hardware and prior to consolidation) is 0.08 m^3 for the reference BWR assembly and 0.18 m^3 for the reference PWR assembly. Assuming two PWRs to every BWR, approximate "untreated" (less hardware and prior to consolidation) concentrations are listed in Table A-73.

A.10 SUMMARY OF WASTE STREAMS AND CONCENTRATIONS

This chapter summarizes "as-generated" (untreated) radionuclide solubilities and concentrations (or activities for source waste streams) considered further in the computer codes for each of the 148 waste streams included in this report. Two principal considerations are of note.

First, the solubility classes assumed for the waste stream radionuclides are listed in Table A-74. The solubility classes for most radionuclides were assumed to be consistent with the assumptions in Appendix B of reference 2. For radionuclides not addressed in reference 2, solubility classes were assigned based on an assumption that the radionuclide chemical form was an oxide. Note that the structure of the analysis methodology allows consideration of radionuclides having different solubility classes within the same waste stream.

Second, Chapter A.9 presents a series of tables, each of which contains projected concentrations (or activities) of a number of radionuclides. Many of these radionuclides have very short half-lives, and are thus not of special concern in terms of long-term impacts from waste disposal. These radionuclides were often not among the 53 radionuclides considered in the computer codes. These radionuclides were included, however, in Chapter A.9 for a number of reasons. One reason is merely a desire to retain collected information for future possible use. The data base could possibly be applied to a number of uses, and retaining accumulated data increases its flexibility. Another reason is that while the current data base emphasizes higher activity waste streams and longer-lived radionuclides, the data base is readily expandable. Future versions may contain a more balanced presentation of waste streams of all activities.

Another reason is that retention of shorter-lived radionuclides allows more accurate calculation of the classification status of distributed waste streams. This is because the concentration distributions were frequently determined by considering gross container concentrations as obtained from shipment records. It also allows for easier comparison of waste projections with actual shipment records.

However, the current version of the updated Part 61 analysis methodology, as discussed above, is mainly concerned with higher activity waste streams and longer-lived radionuclides. In addition, for economic and other reasons, the current version is limited as to the number of radionuclides that can be considered. The solution in this report is to list the concentrations (or activities) of the 53 radionuclides considered in detail in this report within each

Table A-72. Radionuclide Inventory in Reference Spent Fuel Assemblies

Radionuclide	Inventory (Ci/MTHM)	Radionuclide	Inventory (Ci/MTHM)
H-3	4.2E+2	U-234	2.1E-2
C-14	8.0E-1	U-235	1.6E-2
Fe-55	9.0E+1	U-236	2.2E-1
Co-60	1.0E+2	U-238	3.2E-1
Ni-59	3.0E-2	Np-237	3.1E-1
Ni-63	4.0E+0	Pu-236	2.3E-1
Sr-90	6.5E+4	Pu-238	2.1E+3
Tc-99	1.3E+1	Pu-239	2.9E+2
Ru-103	7.2E+1	Pu-240	4.5E+2
Ru-106	1.7E+5	Pu-241	1.1E+5
Ag-110m	6.6E+2	Pu-242	1.6E+0
Sn-126	4.8E-1	Am-241	3.7E+2
I-129	3.3E-2	Am-243	1.4E+1
Cs-135	2.7E-1	Cm-242	3.6E+3
Cs-137	9.2E+4	Cm-243	3.9E+0
Eu-152	1.2E+1	Cm-244	1.3E+3
Eu-154	5.5E+3	Other	1.0E+6
Eu-155	3.9E+3	Total	1.5E+6

Table A-73. Radionuclide Concentration in L-SPENTFU Waste Stream

Radionuclide	Concentration (Ci/m^3)	Radionuclide	Concentration (Ci/m^3)
H-3	1.04E+3	U-234	5.19E-2
C-14	1.98E+0	U-235	3.95E-2
Fe-55	2.22E+2	U-236	5.43E-1
Co-60	2.47E+2	U-238	7.90E-1
Ni-59	7.41E-2	Np-237	7.66E-1
Ni-63	9.88E+0	Pu-236	5.68E-1
Sr-90	1.61E+5	Pu-238	5.19E+3
Tc-99	3.21E+1	Pu-239	7.16E+2
Ru-103	1.78E+2	Pu-240	1.11E+3
Ru-106	4.20E+5	Pu-241	2.72E+5
Ag-110m	1.63E+3	Pu-242	3.95E+0
Sn-126	1.19E+0	Am-241	9.14E+2
I-129	8.15E-2	Am-243	3.46E+1
Cs-135	6.67E-1	Cm-242	8.89E+3
Cs-137	2.27E+5	Cm-243	9.63E+0
Eu-152	2.96E+1	Cm-244	3.21E+3
Eu-154	1.36E+4	Other	2.47E+6
Eu-155	9.63E+3	Total	3.59E+6

Table A-74. Assumed Solubility Classes for Radionuclides Within Waste Streams

Radionuclide	Solubility Class	Radionuclide	Solubility Class
H-3	* (1)	Th-228	Y
C-14	*	Th-229	Y
Na-22	D	Th-230	Y
Cl-36	W	Th-232	Y
Fe-55	Y	Pa-231	W
Co-60	Y	U-232	Y
Ni-59	W	U-233	Y
Ni-63	W	U-234	Y
Sr-90	D	U-235	Y
Nb-94	Y	U-236	Y
Tc-99	D	U-238	Y
Ru-106	Y	Np-237	W
Ag-108m	Y	Pu-236	Y
Cd-109	Y	Pu-238	Y
Sn-126	W	Pu-239	Y
Sb-125	W	Pu-240	Y
I-129	D	Pu-241	Y
Cs-134	D	Pu-242	Y
Cs-135	D	Pu-244	Y
Cs-137	D	Am-241	W
Eu-152	W	Am-243	W
Eu-154	W	Cm-242	W
Pb-210	W	Cm-243	W
Rn-222	*	Cm-244	W
Ra-226	W	Cm-248	W
Ra-228	W	Cf-252	Y
Ac-227	W		

(1) Solubility class: * = not applicable, D = days, W = weeks, Y = year

of the waste streams. Also listed, when information is available, are the total gross concentrations of all radionuclides other than the 53 principal ones. The total radionuclide concentration is also listed.

The radionuclide concentrations or activities, and solubility classes, for all 148 waste streams are listed in Table A-75.

TABLE A-75 . List of Waste Streams and Radionuclides

```
P-IXRESIN - REGULAR WASTE STREAM
 H-3     * 6.13E-01 C-14    * 2.25E-02 FE-55   Y 5.39E-01 CO-60   Y 1.04E+00
 NI-59   W 6.43E-04 NI-63   W 1.98E-01 SR-90   D 4.47E-02 NB-94   Y 2.04E-05
 TC-99   D 1.90E-04 I-129   D 5.62E-04 CS-135  D 1.90E-04 CS-137  D 5.05E+00
 U-235   Y 1.09E-05 U-238   Y 8.55E-05 NP-237  W 2.09E-09 PU-238  Y 5.99E-03
 PU-239  Y 4.20E-03 PU-241  Y 1.83E-01 PU-242  Y 9.20E-06 AM-241  W 4.31E-03
 AM-243  W 2.90E-04 CM-243  W 2.29E-06 CM-244  W 3.18E-03

P-CONCLIQ - REGULAR WASTE STREAM
 H-3     * 2.66E-02 C-14    * 9.80E-04 FE-55   Y 1.75E-01 CO-60   Y 3.40E-01
 NI-59   W 2.09E-04 NI-63   W 6.45E-02 SR-90   D 1.94E-03 NB-94   Y 6.62E-06
 TC-99   D 8.26E-06 I-129   D 2.44E-05 CS-135  D 8.26E-06 CS-137  D 2.20E-01
 U-235   Y 4.75E-07 U-238   Y 3.73E-06 NP-237  W 9.10E-11 PU-238  Y 3.95E-04
 PU-239  Y 2.55E-04 PU-241  Y 1.11E-02 PU-242  Y 5.59E-07 AM-241  W 2.31E-04
 AM-243  W 1.56E-05 CM-243  W 9.03E-08 CM-244  W 1.48E-04

P-FSLUDGE - REGULAR WASTE STREAM
 H-3     * 1.89E-02 C-14    * 6.97E-04 FE-55   Y 2.26E+00 CO-60   Y 4.38E+00
 NI-59   W 2.71E-03 NI-63   W 8.33E-01 SR-90   D 1.38E-03 NB-94   Y 8.55E-05
 TC-99   D 5.86E-06 I-129   D 1.73E-05 CS-135  D 5.86E-06 CS-137  D 1.56E-01
 U-235   Y 1.07E-06 U-238   Y 8.40E-06 NP-237  W 2.05E-10 PU-238  Y 3.48E-04
 PU-239  Y 1.13E-03 PU-241  Y 4.93E-02 PU-242  Y 2.48E-06 AM-241  W 1.93E-03
 AM-243  W 1.30E-04 CM-243  W 2.26E-06 CM-244  W 1.29E-03

P-FCARTRG - REGULAR WASTE STREAM
 H-3     * 2.77E-03 C-14    * 1.02E-04 FE-55   Y 1.34E+00 CO-60   Y 2.58E+00
 NI-59   W 1.59E-03 NI-63   W 4.91E-01 SR-90   D 2.02E-04 NB-94   Y 5.03E-05
 TC-99   D 8.62E-07 I-129   D 2.55E-06 CS-135  D 8.62E-07 CS-137  D 2.30E-02
 U-235   Y 8.77E-07 U-238   Y 6.91E-06 NP-237  W 1.69E-10 PU-238  Y 6.05E-04
 PU-239  Y 9.15E-04 PU-241  Y 4.00E-02 PU-242  Y 2.01E-06 AM-241  W 3.95E-04
 AM-243  W 2.65E-05 CM-243  W 4.65E-07 CM-244  W 2.65E-04

B-IXRESIN - REGULAR WASTE STREAM
 H-3     * 2.32E-02 C-14    * 1.44E-03 FE-55   Y 1.15E+00 CO-60   Y 1.92E+00
 NI-59   W 1.18E-03 NI-63   W 2.60E-02 SR-90   D 4.40E-03 NB-94   Y 3.74E-05
 TC-99   D 9.25E-05 I-129   D 2.47E-04 CS-135  D 9.25E-05 CS-137  D 2.47E+00
 U-235   Y 6.44E-08 U-238   Y 5.08E-07 NP-237  W 1.23E-11 PU-238  Y 1.01E-04
 PU-239  Y 6.46E-05 PU-241  Y 3.14E-03 PU-242  Y 1.41E-07 AM-241  W 2.80E-05
 AM-243  W 1.90E-06 CM-243  W 3.26E-08 CM-244  W 2.20E-05

B-CONCLIQ - REGULAR WASTE STREAM
 H-3     * 1.95E-03 C-14    * 1.22E-04 FE-55   Y 2.38E-01 CO-60   Y 3.97E-01
 NI-59   W 2.45E-04 NI-63   W 5.38E-03 SR-90   D 3.69E-04 NB-94   Y 7.75E-06
 TC-99   D 7.82E-06 I-129   D 2.08E-05 CS-135  D 7.82E-06 CS-137  D 2.08E-01
 U-235   Y 1.08E-07 U-238   Y 8.47E-07 NP-237  W 2.07E-11 PU-238  Y 6.22E-04
 PU-239  Y 2.95E-04 PU-241  Y 1.44E-02 PU-242  Y 6.14E-07 AM-241  W 3.75E-04
 AM-243  W 2.53E-05 CM-243  W 8.10E-07 CM-244  W 6.41E-04
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
B-FSLUDGE - REGULAR WASTE STREAM
 H-3    * 1.35E-02 C-14    * 8.32E-04 FE-55  Y 1.54E+00 CO-60  Y 2.58E+00
 NI-59  W 1.59E-03 NI-63   W 3.47E-02 SR-90  D 2.53E-03 NB-94  Y 5.02E-05
 TC-99  D 5.35E-05 I-129   D 1.42E-04 CS-135 D 5.35E-05 CS-137 D 1.42E+00
 U-235  Y 3.55E-07 U-238   Y 2.79E-06 NP-237 W 6.82E-11 PU-238 Y 4.98E-04
 PU-239 Y 2.52E-04 PU-241  Y 1.23E-02 PU-242 Y 5.54E-07 AM-241 W 1.67E-04
 AM-243 W 1.12E-05 CM-243  W 3.17E-07 CM-244 W 2.39E-04

P-COTRASH - REGULAR WASTE STREAM
 H-3    * 7.32E-04 C-14    * 2.70E-05 FE-55  Y 1.44E-02 CO-60  Y 2.78E-02
 NI-59  W 1.72E-05 NI-63   W 5.29E-03 SR-90  D 5.34E-05 NB-94  Y 5.45E-07
 TC-99  D 2.28E-07 I-129   D 6.73E-07 CS-135 D 2.28E-07 CS-137 D 6.04E-03
 U-235  Y 1.91E-08 U-238   Y 1.50E-07 NP-237 W 3.67E-12 PU-238 Y 1.44E-05
 PU-239 Y 1.34E-05 PU-241  Y 5.82E-04 PU-242 Y 2.92E-08 AM-241 W 9.56E-06
 AM-243 W 6.47E-09 CM-243  W 6.63E-09 CM-244 W 6.31E-06

P-NCTRASH - REGULAR WASTE STREAM
 H-3    * 5.03E-03 C-14    * 1.85E-04 FE-55  Y 9.86E-02 CO-60  Y 1.91E-01
 NI-59  W 1.18E-04 NI-63   W 3.63E-02 SR-90  D 3.68E-04 NB-94  Y 3.73E-06
 TC-99  D 1.56E-06 I-129   D 4.61E-06 CS-135 D 1.56E-06 CS-137 D 4.15E-02
 U-235  Y 1.31E-07 U-238   Y 1.03E-06 NP-237 W 2.52E-11 PU-238 Y 9.92E-05
 PU-239 Y 9.14E-05 PU-241  Y 3.99E-03 PU-242 Y 2.01E-07 AM-241 W 6.55E-05
 AM-243 W 4.43E-06 CM-243  W 4.58E-08 CM-244 W 4.31E-05

B-COTRASH - REGULAR WASTE STREAM
 H-3    * 5.67E-05 C-14    * 3.50E-06 FE-55  Y 5.03E-03 CO-60  Y 8.47E-03
 NI-59  W 5.21E-06 NI-63   W 1.14E-04 SR-90  D 1.06E-05 NB-94  Y 1.64E-07
 TC-99  D 2.25E-07 I-129   D 5.99E-07 CS-135 D 2.25E-07 CS-137 D 5.99E-03
 U-235  Y 1.02E-09 U-238   Y 8.04E-09 NP-237 W 1.97E-13 PU-238 Y 1.92E-06
 PU-239 Y 9.72E-07 PU-241  Y 4.72E-05 PU-242 Y 2.12E-09 AM-241 W 8.11E-07
 AM-243 W 5.47E-08 CM-243  W 1.62E-09 CM-244 W 1.26E-06

B-NCTRASH - REGULAR WASTE STREAM
 H-3    * 2.46E-04 C-14    * 1.51E-05 FE-55  Y 2.18E-02 CO-60  Y 3.65E-02
 NI-59  W 2.26E-05 NI-63   W 4.92E-04 SR-90  D 4.63E-05 NB-94  Y 7.11E-07
 TC-99  D 9.38E-07 I-129   D 2.59E-06 CS-135 D 9.76E-07 CS-137 D 2.59E-02
 U-235  Y 4.46E-09 U-238   Y 3.49E-08 NP-237 W 8.50E-13 PU-238 Y 8.37E-06
 PU-239 Y 4.19E-06 PU-241  Y 2.04E-04 PU-242 Y 9.21E-09 AM-241 W 3.52E-06
 AM-243 W 2.36E-07 CM-243  W 7.03E-09 CM-244 W 5.43E-06

L-NFRCOMP - ACTIVATED METAL WASTE STREAM
 H-3    * 3.34E+00 C-14    * 1.91E-03 CL-36  W 2.38E-04 FE-55  Y 4.29E+01
 CO-60  Y 4.10E+01 NI-59   W 1.71E-01 NI-63  W 5.18E+00 SR-90  D 1.57E-04
 NB-94  Y 4.06E-05 TC-99   D 7.03E-05 CS-137 D 5.09E-04            3.01E+00

L-DECONRS - REGULAR WASTE STREAM
 FE-55  Y 2.63E+00 CO-60   Y 1.89E+01 NI-63  W 9.96E-01 RU-106 Y 8.46E-01
 SB-125 W 1.88E-03 EU-154  W 3.76E-05 PU-238 Y 1.13E-02 PU-239 Y 7.52E-03
 CM-242 W 1.13E-02 CM-244  W 3.76E-03          1.59E+00
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
F-PROCESS - REGULAR WASTE STREAM
 U-235  Y 2.30E-05 U-238  Y 8.54E-05

F-COTRASH - REGULAR WASTE STREAM
 U-235  Y 1.18E-06 U-238  Y 4.40E-06

F-NCTRASH - REGULAR WASTE STREAM
 U-235  Y 1.13E-06 U-238  Y 4.20E-06

U-PROCESS - REGULAR WASTE STREAM
 U-235  Y 1.65E-05 U-238  Y 3.64E-04

L-PUDECON - REGULAR WASTE STREAM
 PU-238 Y 8.13E-02 PU-239 Y 1.28E-01 PU-241 Y 3.85E+00 PU-242 Y 4.80E-05
 AM-241 W 1.11E-01

L-BURNUPS - REGULAR WASTE STREAM
 H-3    * 4.16E+01 C-14   * 7.93E-02 FE-55  Y 8.92E+00 CO-60  Y 9.91E+00
 NI-59  W 2.97E-03 NI-63  W 3.96E-01 SR-90  D 6.44E+03 TC-99  D 1.29E+00
 RU-106 Y 1.68E+04 CM-244 W 1.29E+02 SN-126 W 4.76E-02 I-129  D 3.27E-03
 CS-135 D 2.68E-02 CS-137 D 9.12E+03 EU-152 W 1.19E+00 EU-154 W 5.45E+02
 U-234  Y 2.08E-03 U-235  Y 1.59E-03 U-236  Y 2.18E-02 U-238  Y 3.17E-02
 NP-237 W 3.07E-02 PU-236 Y 2.28E-02 PU-238 Y 2.08E+02 PU-239 Y 7.33E+01
 PU-241 Y 1.09E+04 PU-242 Y 1.59E-01 AM-241 W 3.67E+01 AM-243 W 1.39E+00
 CM-242 W 3.57E+02 CM-243 W 3.86E-01          6.90E+02

I-COTRASH - REGULAR WASTE STREAM
 H-3    * 9.13E-02 C-14   * 5.26E-03 CO-60  Y 1.04E-02 SR-90  D 1.45E-03
 TC-99  D 3.39E-09 CS-137 D 4.56E-03 AM-241 W 4.82E-06

I+COTRASH - REGULAR WASTE STREAM
 H-3    * 9.13E-02 C-14   * 5.26E-03 CO-60  Y 1.04E-02 SR-90  D 1.45E-03
 TC-99  D 3.39E-09 CS-137 D 4.55E-03 AM-241 W 4.82E-06

I-ABSLIQD - REGULAR WASTE STREAM
 H-3    * 1.42E-01 C-14   * 8.16E-03 CO-60  Y 3.12E-02 SR-90  D 4.34E-03
 TC-99  D 1.02E-08 CS-137 D 1.37E-02

I+ABSLIQD - REGULAR WASTE STREAM
 H-3    * 1.42E-01 C-14   * 8.16E-03 CO-60  Y 3.12E-02 SR-90  D 4.34E-03
 TC-99  D 1.02E-08 CS-137 D 1.37E-02

I-LIQSCVL - REGULAR WASTE STREAM
 H-3    * 5.01E-03 C-14   * 2.51E-04 CO-60  Y 4.34E-03

I+LIQSCVL - REGULAR WASTE STREAM
 H-3    * 5.01E-03 C-14   * 2.51E-04 CO-60  Y 4.34E-03
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
I-BIOWAST - REGULAR WASTE STREAM
 H-3    * 1.75E-01 C-14   * 1.01E-02 CO-60  Y 3.99E-03 SR-90  D 8.33E-03
 TC-99  D 6.51E-09 CS-137 D 8.76E-03

I+BIOWAST - REGULAR WASTE STREAM
 H-3    * 1.75E-01 C-14   * 1.01E-02 CO-60  Y 3.99E-03 SR-90  D 8.33E-03
 TC-99  D 6.51E-09 CS-137 D 8.76E-03

N-SSTRASH - REGULAR WASTE STREAM
 U-235  Y 2.36E-06 U-238  Y 8.80E-06

N+SSTRASH - REGULAR WASTE STREAM
 U-235  Y 2.36E-06 U-238  Y 8.80E-06

N-SSWASTE - REGULAR WASTE STREAM
 U-235  Y 4.60E-05 U-238  Y 1.71E-04

N-LOTRASH - REGULAR WASTE STREAM
 H-3    * 2.85E-02 C-14   * 1.64E-03 CO-60  Y 3.25E-03 SR-90  D 4.53E-04
 TC-99  D 1.06E-09 CS-137 D 1.42E-03 AM-241 W 1.51E-06

N+LOTRASH - REGULAR WASTE STREAM
 H-3    * 2.85E-02 C-14   * 1.64E-03 CO-60  Y 3.25E-03 SR-90  D 4.53E-04
 TC-99  D 1.06E-09 CS-137 D 1.42E-03 AM-241 W 1.51E-06

N-LOWASTE - REGULAR WASTE STREAM
 H-3    * 1.63E-02 C-14   * 9.36E-04 CO-60  Y 1.47E-03 SR-90  D 1.31E-03
 TC-99  D 7.76E-10 CS-137 D 1.04E-03

N-ISOPROD - REGULAR WASTE STREAM
 H-3    * 1.09E-02 C-14   * 1.17E-05 FE-55  Y 1.60E+01 NI-63  W 2.46E-01
 SR-90  D 2.35E+00 TC-99  D 8.48E-05 RU-106 Y 2.42E+00 I-129  D 7.04E-07
 CS-135 D 8.48E-05 CS-137 D 1.54E+00 U-235  Y 4.70E-04 U-238  Y 5.48E-06
 NP-237 W 1.03E-13 PU-238 Y 3.81E-05 PU-239 Y 1.07E-05 PU-241 Y 1.37E-03
 PU-242 Y 1.85E-08 AM-241 W 2.12E-06 AM-243 W 2.43E-07 CM-242 W 2.27E-04
 CM-243 W 5.57E-08 CM-244 W 3.20E-05          2.22E+02

N-ISOTRSH - REGULAR WASTE STREAM
 H-3    * 1.51E-06 C-14   * 1.62E-09 CO-60  Y 5.63E-04 SR-90  D 2.70E-04
 TC-99  D 1.18E-08 CM-243 W 1.79E-11 CM-244 W 1.03E-08 I-129  D 9.77E-11
 CS-135 D 1.18E-08 CS-137 D 2.70E-04 U-235  Y 3.00E-05 U-238  Y 1.67E-07
 NP-237 W 3.31E-17 PU-238 Y 1.22E-08 PU-239 Y 3.44E-09 PU-241 Y 4.40E-07
 PU-242 Y 5.94E-12 AM-241 W 6.80E-10 AM-243 W 7.77E-11 CM-242 W 7.29E-08

N-SORMFG1 - REGULAR WASTE STREAM
 AM-241 W 8.00E+00

N-SORMFG2 - REGULAR WASTE STREAM
 CO-60  Y 2.90E+00 CS-137 D 6.50E-01
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
N-SORMFG3 - REGULAR WASTE STREAM
 CS-137 D 6.00E+03

N-SORMFG4 - REGULAR WASTE STREAM
 H-3    * 1.70E-01 C-14    * 2.40E-04 SR-90  D 2.63E-03 CS-137 D 4.31E-01
 TH-228 Y 4.54E-09           5.03E-01

N-NECOTRA - REGULAR WASTE STREAM
 H-3    * 3.80E-01 C-14    * 4.07E-02          1.19E-01

N-NEABLIQ - REGULAR WASTE STREAM
 H-3    * 5.48E+01 C-14    * 8.43E-01          1.89E+01

N-NESOLIQ - REGULAR WASTE STREAM
 H-3    * 2.60E+01 C-14    * 1.56E-01          8.57E-02

N-NEVIALS - REGULAR WASTE STREAM
 H-3    * 3.09E+01 C-14    * 7.37E-01          1.59E+01

N-NENCGLS - REGULAR WASTE STREAM
 H-3    * 1.99E+01 C-14    * 1.88E-02          3.57E-02

N-NEWOTAL - REGULAR WASTE STREAM
 H-3    * 4.15E-01 C-14    * 1.51E-01          4.98E-02

N-NETRGAS - REGULAR WASTE STREAM
 H-3    * 4.62E+04

N-NETRILI - REGULAR WASTE STREAM
 H-3    * 4.62E+04

N-NECARLI - REGULAR WASTE STREAM
 C-14   * 4.11E+02

N-MWTRASH - REGULAR WASTE STREAM
 H-3    * 1.59E+00 C-14    * 2.48E-01

N-MWABLIQ - REGULAR WASTE STREAM
 H-3    * 1.93E+02 C-14    * 1.40E+01

N-MWSOLIQ - REGULAR WASTE STREAM
 H-3    * 1.01E+03

N-MWWASTE - REGULAR WASTE STREAM
 H-3    * 6.69E+00 C-14    * 2.83E+01

N-TRIPLAT - REGULAR WASTE STREAM
 H-3    * 4.89E+02
```

TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
N-TRITGAS - REGULAR WASTE STREAM
 H-3    * 5.94E+02

N-TRISCNT - REGULAR WASTE STREAM
 H-3    * 4.22E+01

N-TRILIQD - REGULAR WASTE STREAM
 H-3    * 7.11E+02

N-TRITRSH - REGULAR WASTE STREAM
 H-3    * 6.94E+01

N-TRIFOIL - REGULAR WASTE STREAM
 H-3    * 7.80E+02

N-HIGHACT - ACTIVATED METAL WASTE STREAM
 C-14   * 1.32E-02 FE-55  Y 1.15E+02 CO-60  Y 8.48E+01 NI-59  W 6.56E-02
 NI-63  W 1.06E+01 NB-94  Y 4.47E-04

N-TRITSOR - SOURCE WASTE STREAM
 H-3    * 1.00E+00

N-CARBSOR - SOURCE WASTE STREAM
 C-14   * 1.00E-02

N-COBLSOR - SOURCE WASTE STREAM
 CO-60  Y 5.00E+02

N-NICKSOR - SOURCE WASTE STREAM
 NI-63  W 1.00E-02

N-STROSOR - SOURCE WASTE STREAM
 SR-90  D 1.00E+00

N-CESISOR - SOURCE WASTE STREAM
 CS-137 D 1.00E+02

N-PLU8SOR - SOURCE WASTE STREAM
 PU-238 Y 4.60E+00

N-PLU9SOR - SOURCE WASTE STREAM
 PU-239 Y 3.00E+00

N-AMERSOR - SOURCE WASTE STREAM
 AM-241 W 5.00E-01

N-PUBESOR - SOURCE WASTE STREAM
 PU-238 Y 2.35E+01
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
N-AMBESOR - SOURCE WASTE STREAM
 AM-241 W 8.20E+00

N-RANEEDS - SOURCE WASTE STREAM
 RA-226 W 5.65E-03

N-RACELLS - SOURCE WASTE STREAM
 RA-226 W 7.09E-03

N-RAPLAQU - SOURCE WASTE STREAM
 RA-226 W 1.00E-02

N-RANPAPP - SOURCE WASTE STREAM
 RA-226 W 5.00E-02

N-RABESOR - SOURCE WASTE STREAM
 RA-226 W 4.57E-01

N-RAMISCL - SOURCE WASTE STREAM
 RA-226 W 2.27E-02

N-RARESIN - REGULAR WASTE STREAM
 RA-226 W 3.50E-02

M-NAVYDRY - REGULAR WASTE STREAM
 H-3    * 2.67E-04 C-14   * 9.82E-06 FE-55  Y 5.24E-03 CO-60  Y 1.01E-02
 NI-59  W 6.24E-06 NI-63  W 1.92E-03 SR-90  D 1.95E-05 NB-94  Y 1.97E-07
 TC-99  D 8.26E-08 I-129  D 2.44E-07 CS-135 D 8.26E-08 CS-137 D 2.20E-03
 U-235  Y 6.92E-09 U-238  Y 5.46E-08 NP-237 W 1.33E-12 PU-238 Y 5.24E-06
 PU-239 Y 4.85E-06 PU-241 Y 2.11E-04 PU-242 Y 1.06E-08 AM-241 W 3.47E-06
 AM-243 W 2.34E-07 CM-243 W 2.40E-09 CM-244 W 2.29E 06

M-NAVYWET - REGULAR WASTE STREAM
 H-3    * 6.26E-02 C-14   * 2.29E-03 FE-55  Y 5.51E-02 CO-60  Y 1.07E-01
 NI-59  W 6.57E-05 NI-63  W 2.03E-02 SR-90  D 4.57E-03 NB-94  Y 2.08E-06
 TC-99  D 1.94E-05 I-129  D 5.73E-05 CS-135 D 1.94E-05 CS-137 D 5.16E-01
 U-235  Y 1.11E-06 U-238  Y 8.74E-06 NP-237 W 2.13E-10 PU-238 Y 6.12E-04
 PU-239 Y 4.29E-04 PU-241 Y 1.87E-02 PU-242 Y 9.40E-07 AM-241 W 4.40E-04
 AM-243 W 2.97E-05 CM-243 W 2.34E-07 CM-244 W 3.25E-04

R-HLLWFRP - REGULAR WASTE STREAM
 H-3    * 5.60E+01 C-14   * 9.33E-01 SR-90  D 1.02E+05 TC-99  D 2.17E+01
 RU-106 Y 3.17E+05 SN-126 W 9.17E-01 SB-125 W 9.83E+03 I-129  D 2.92E-04
 CS-134 D 2.00E+05 CS-135 D 5.17E-01 CS-137 D 1.55E+05 EU-152 W 2.67E+01
 EU-154 W 1.02E+04 TH-228 Y 8.83E-03 TH-230 Y 2.13E-05 PA-231 W 1.78E-05
 U-232  Y 9.93E-05 U-233  Y 3.25E-07 U-234  Y 2.80E-03 U-235  Y 1.33E-04
 U-236  Y 2.20E-03 U-238  Y 2.62E-03 PU-236 Y 2.68E-03 PU-238 Y 4.62E+01
 PU-239 Y 2.98E+00 PU-240 Y 6.10E+00 PU-241 Y 1.46E+03 PU-242 Y 3.25E-02
 AM-241 W 1.18E+03 AM-243 W 7.88E+01 CM-242 W 1.65E+04 CM-243 W 1.62E+01
 CM-244 W 1.19E+04 CM-248 W 1.24E-05 CF-252 Y 5.62E-04 AC-227 W 1.61E-06
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

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R-FUEHARD - ACTIVATED METAL WASTE STREAM
 C-14    * 8.90E-01 FE-55   Y 7.10E+04 CO-60   Y 7.10E+04 NI-59   W 5.40E+01
 NI-63   W 7.40E+03 SR-90   D 1.80E-02 NB-94   Y 1.80E-03 TC-99   D 1.30E-01

R-HULLFRP - ACTIVATED METAL WASTE STREAM
 H-3     * 2.40E+02 C-14    * 2.30E-01 FE-55   Y 3.40E+02 CO-60   Y 3.80E+02
 NI-59   W 1.10E-01 NI-63   W 1.50E+01 SR-90   D 1.20E+02 TC-99   D 2.40E-02
 CS-135  D 5.80E-04 CS-137  D 1.80E+02 U-234   Y 6.40E-04 U-235   Y 3.00E-05
 U-236   Y 4.90E-04 U-238   Y 6.00E-04 NP-237  W 7.50E-04 PU-236  Y 6.00E-04
 PU-238  Y 1.00E+01 PU-239  Y 6.80E-01 PU-240  Y 1.40E+00 PU-241  Y 3.40E+02
 PU-242  Y 7.30E-03 AM-241  W 1.30E+00 AM-243  W 8.80E-02 CM-242  W 1.90E+01
 CM-243  W 1.80E-02 CM-244  W 1.40E+01

R-ILLWFRP - REGULAR WASTE STREAM
 H-3     * 3.80E+00 C-14    * 5.10E-05 SR-90   D 5.60E+00 TC-99   D 1.30E-04
 I-129   D 9.60E-04 CS-135  D 2.80E-05 CS-137  D 8.50E+00 U-234   Y 3.10E-03
 U-235   Y 1.50E-04 U-236   Y 2.40E-03 U-238   Y 2.90E-03 NP-237  W 3.60E-05
 PU-236  Y 2.90E-03 PU-238  Y 5.10E+01 PU-239  Y 3.30E+00 PU-240  Y 6.60E+00
 PU-241  Y 1.60E+03 PU-242  Y 3.60E-02 AM-241  W 6.50E-02 AM-243  W 4.50E-03
 CM-242  W 9.00E-01 CM-243  W 8.80E-04 CM-244  W 6.60E-01

R-SILIGEL - REGULAR WASTE STREAM
 PU-236  Y 6.40E-06 PU-238  Y 1.10E-01 PU-239  Y 7.20E-03 PU-240  Y 1.50E-02
 PU-241  Y 3.60E+00 PU-242  Y 7.80E-05

R-MPCOTRH - REGULAR WASTE STREAM
 C-14    * 1.53E-06 SR-90   D 1.65E-01 TC-99   D 3.61E-05 CS-135  D 8.54E-07
 CS-137  D 2.52E-01 U-234   Y 9.31E-07 U-235   Y 4.37E-08 U-236   Y 7.14E-07
 U-238   Y 8.77E-07 NP-237  W 1.10E-06 PU-236  Y 8.77E-07 PU-238  Y 1.53E-02
 PU-239  Y 9.88E-04 PU-240  Y 1.98E-03 PU-241  Y 4.94E-01 PU-242  Y 1.07E-05
 AM-241  W 1.97E-03 AM-243  W 1.31E-04 CM-242  W 2.74E-02 CM-243  W 2.63E-05
 CM-244  W 1.97E-02

R-MPCOTRL - REGULAR WASTE STREAM
 C-14    * 4.70E-10 SR-90   D 5.10E-05 TC-99   D 1.10E-08 CS-135  D 2.60E-10
 CS-137  D 7.80E-05 U-234   Y 2.80E-10 U-235   Y 1.30E-11 U-236   Y 2.20E-10
 U-238   Y 2.70E-10 NP-237  W 3.30E-10 PU-236  Y 2.70E-10 PU-238  Y 4.60E-06
 PU-239  Y 3.00E-07 PU-240  Y 6.10E-07 PU-241  Y 1.50E-04 PU-242  Y 3.30E-09
 AM-241  W 5.90E-07 AM-243  W 3.90E-08 CM-242  W 8.30E-06 CM-243  W 8.10E-09
 CM-244  W 6.00E-06

R-MPNCTRA - REGULAR WASTE STREAM
 C-14    * 1.87E-06 SR-90   D 2.03E-01 TC-99   D 4.37E-05 CS-135  D 1.05E-06
 CS-137  D 3.10E-01 U-234   Y 1.13E-06 U-235   Y 5.33E-08 U-236   Y 8.67E-07
 U-238   Y 1.07E-06 NP-237  W 1.33E-06 PU-236  Y 1.07E-06 PU-238  Y 1.87E-02
 PU-239  Y 1.20E-03 PU-240  Y 2.40E-03 PU-241  Y 6.00E-01 PU-242  Y 1.32E-05
 AM-241  W 1.32E-03 AM-243  W 1.60E-04 CM-242  W 3.33E-02 CM-243  W 3.23E-05
 CM-244  W 2.40E-02
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
R-DEGREXT - REGULAR WASTE STREAM
 C-14   * 7.00E-09 SR-90  D 7.60E-04 TC-99  D 1.60E-07 I-129  D 4.40E-04
 CS-135 D 3.90E-09 CS-137 D 1.20E-03 U-234  Y 4.30E-05 U-235  Y 2.00E-06
 U-236  Y 3.30E-05 U-238  Y 4.00E-05 NP-237 W 5.00E-05 PU-236 Y 4.00E-03
 PU-238 Y 6.90E+01 PU-239 Y 4.50E+00 PU-240 Y 9.10E+00 PU-241 Y 2.30E+03
 PU-242 Y 4.90E-02 AM-241 W 8.90E-02 AM-243 W 5.90E-03 CM-242 W 1.20E+00
 CM-243 W 1.20E+00 CM-244 W 9.00E-01

R-MPRESIN - REGULAR WASTE STREAM
 C-14   * 1.10E-05 SR-90  D 1.20E+00 TC-99  D 2.60E-04 I-129  D 1.40E-02
 CS-135 D 6.20E-06 CS-137 D 1.90E+00 U-234  Y 6.80E-04 U-235  Y 3.20E-05
 U-236  Y 5.20E-04 U-238  Y 6.40E-04 NP-237 W 8.00E-06 PU-236 Y 6.40E-04
 PU-238 Y 1.10E+01 PU-239 Y 7.20E-01 PU-240 Y 1.50E+00 PU-241 Y 3.60E+02
 PU-242 Y 7.80E-03 AM-241 W 1.40E-02 AM-243 W 9.40E-04 CM-242 W 2.00E-01
 CM-243 W 1.90E-04 CM-244 W 1.40E-01

R-SBRESIN - REGULAR WASTE STREAM
 FE-55  Y 7.78E+00 CO-60  Y 4.86E+00 SR-90  D 6.21E-01 CS-137 D 3.69E+01

R-SBCOLIQ - REGULAR WASTE STREAM
 FE-55  Y 1.08E+01 CO-60  Y 6.77E+00 SR-90  D 8.60E-01 CS-137 D 5.13E+01

R-SBCOTRA - REGULAR WASTE STREAM
 FE-55  Y 5.28E-03 CO-60  Y 3.31E-03 SR-90  D 4.30E-04 CS-137 D 2.54E-02

R-SBNCTRA - REGULAR WASTE STREAM
 FE-55  Y 2.36E-03 CO-60  Y 1.44E-03 SR-90  D 1.86E-04 CS-137 D 1.10E-02

R-UFFINES - REGULAR WASTE STREAM
 U-234  Y 1.51E-02 U-235  Y 7.09E-04 U-236  Y 1.15E-02 U-238  Y 1.42E-02
 PU-236 Y 1.16E-05 PU-238 Y 1.92E-01 PU-239 Y 1.27E-02 PU-240 Y 2.64E-02
 PU-241 Y 6.35E+00 PU-242 Y 1.38E-04

R-UFK2MUD - REGULAR WASTE STREAM
 U-234  Y 6.30E-03 U-235  Y 3.00E-04 U-236  Y 4.80E-03 U-238  Y 5.90E-03

R-UFCOTRA - REGULAR WASTE STREAM
 U-234  Y 3.15E-03 U-235  Y 1.48E-04 U-236  Y 2.41E-03 U-238  Y 2.97E-03

R-UFNCTRA - REGULAR WASTE STREAM
 U-234  Y 1.87E-03 U-235  Y 8.80E-05 U-236  Y 1.43E-03 U-238  Y 1.76E-03

R-PUCOTRA - REGULAR WASTE STREAM
 PU-236 Y 1.65E-02 PU-238 Y 2.75E+02 PU-239 Y 1.80E+01 PU-240 Y 3.64E+01
 PU-241 Y 8.48E+03 PU-242 Y 1.95E-01

R-PUNCTRA - REGULAR WASTE STREAM
 PU-236 Y 2.58E-03 PU-238 Y 4.28E+01 PU-239 Y 2.77E+00 PU-240 Y 5.62E+00
 PU-241 Y 1.30E+03 PU-242 Y 3.04E-02
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

```
R-MOXCOTR - REGULAR WASTE STREAM
 U-234  Y 6.42E-04 U-235  Y 2.58E-05 U-236  Y 1.26E-07 U-238  Y 5.25E-04
 NP-237 W 2.63E-07 PU-236 Y 1.50E-03 PU-238 Y 3.08E+01 PU-239 Y 2.17E+00
 PU-240 Y 4.33E+00 PU-241 Y 9.75E+02 PU-242 Y 2.33E-02 AM-241 W 1.60E+00

R-MOXNCTR - REGULAR WASTE STREAM
 U-234  Y 1.10E-04 U-235  Y 4.13E-06 U-236  Y 2.09E-08 U-238  Y 8.80E-05
 NP-237 W 4.40E-08 PU-236 Y 2.48E-04 PU-238 Y 5.23E+00 PU-239 Y 3.53E-01
 PU-240 Y 7.15E-01 PU-241 Y 1.60E+02 PU-242 Y 3.85E-03 AM-241 W 2.64E-01

R-MOXSOLN - REGULAR WASTE STREAM
 U-234  Y 1.15E-04 U-235  Y 4.07E-06 U-236  Y 2.19E-08 U-238  Y 9.47E-05
 NP-237 W 4.78E-08 PU-236 Y 2.70E-04 PU-238 Y 5.62E+00 PU-239 Y 3.85E-01
 PU-240 Y 7.71E-01 PU-241 Y 1.76E+02 PU-242 Y 4.16E-03 AM-241 W 5.21E+01

P-DECORES - ACTIVATED METAL WASTE STREAM
 H-3    * 8.66E+00 C-14   * 2.17E+01 CL-36  W 4.42E-01 FE-55  Y 1.82E+05
 CO-60  Y 1.12E+05 NI-59  W 9.53E+01 NI-63  W 1.55E+04 SR-90  D 1.73E+00
 NB-94  Y 3.47E-01 TC-99  D 1.12E-01 I-129  D 5.20E-07 CS-135 D 3.47E-05
 CS-137 D 1.73E+00 U-233  Y 3.12E-04 PU-239 Y 6.06E-03            1.22E+00

P-DEACINT - ACTIVATED METAL WASTE STREAM
 H-3    * 1.35E+00 C-14   * 4.23E-01 CL-36  W 9.03E-03 FE-55  Y 3.38E+03
 CO-60  Y 1.97E+03 NI-59  W 2.68E+00 NI-63  W 3.35E+02 SR-90  D 7.05E-04
 NB-94  Y 4.09E-03 TC-99  D 1.16E-03 I-129  D 1.97E-09 CS-135 D 1.27E-07
 CS-137 D 7.05E-03 U-233  Y 1.41E-05 PU-239 Y 3.24E-04            1.00E+01

P-DEACVES - ACTIVATED METAL WASTE STREAM
 H-3    * 1.16E-02 C-14   * 6.53E-03 CL-36  W 1.43E-04 FE-55  Y 5.28E+01
 CO-60  Y 2.87E+01 NI-59  W 3.84E-02 NI-63  W 5.10E+00 SR-90  D 1.34E-04
 NB-94  Y 5.64E-05 TC-99  D 1.34E-05 I-129  D 3.76E-11 CS-135 D 2.42E-09
 CS-137 D 1.34E-04 U-233  Y 1.43E-07 PU-239 Y 2.42E-06            1.67E-04

P-DEACTCO - REGULAR WASTE STREAM
 H-3    * 8.56E-01 C-14   * 3.09E-04 CL-36  W 3.32E-10 FE-55  Y 1.86E+00
 CO-60  Y 7.95E-02 NI-59  W 1.12E-05 NI-63  W 1.41E-03 SR-90  D 7.80E-06
 NB-94  Y 1.11E-06 TC-99  D 4.42E-08 I-129  D 2.36E-12 CS-135 D 1.50E-10
 CS-137 D 8.13E-06 U-233  Y 1.24E-06 PU-239 Y 1.81E-06            2.27E-02

P-DECONME - REGULAR WASTE STREAM
 H-3    * 1.10E-05 C-14   * 5.26E-07 FE-55  Y 4.16E-02 CO-60  Y 7.47E-02
 NI-59  W 4.61E-05 NI-63  W 3.78E-03 SR-90  D 4.38E-05 NB-94  Y 1.46E-06
 TC-99  D 1.23E-08 I-129  D 3.42E-08 CS-135 D 1.23E-08 CS-137 D 3.84E-04
 U-235  Y 7.02E-08 U-238  Y 5.54E-07 NP-237 W 1.35E-11 PU-238 Y 1.37E-03
 PU-239 Y 1.81E-03 PU-241 Y 3.36E-02 PU-242 Y 3.97E-06 AM-241 W 5.42E-06
 AM-243 W 3.68E-07 CM-243 W 3.55E-07 CM-244 W 3.36E-06
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

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P-DECONCO - REGULAR WASTE STREAM
 H-3    * 6.49E-07 C-14    * 3.09E-08 FE-55   Y 2.44E-03 CO-60   Y 4.39E-03
 NI-59  W 2.71E-06 NI-63   W 2.22E-04 SR-90   D 2.58E-06 NB-94   Y 8.56E-08
 TC-99  D 7.21E-10 I-129   D 2.10E-09 CS-135  D 7.21E-10 CS-137  D 2.26E-05
 U-235  Y 4.13E-09 U-238   Y 3.25E-08 NP-237  W 7.93E-13 PU-238  Y 8.07E-05
 PU-239 Y 1.06E-04 PU-241  Y 2.13E-03 PU-242  Y 2.33E-07 AM-241  W 3.19E-07
 AM-243 W 2.16E-08 CM-243  W 2.09E-08 CM-244  W 1.97E-07

P-DETRASH - REGULAR WASTE STREAM
 H-3    * 7.05E-03 C-14    * 2.61E-04 FE-55   Y 1.39E-01 CO-60   Y 2.68E-01
 NI-59  W 1.65E-04 NI-63   W 5.10E-02 SR-90   D 5.15E-04 NB-94   Y 5.51E-06
 TC-99  D 2.19E-06 I-129   D 6.49E-06 CS-135  D 2.19E-06 CS-137  D 5.82E-02
 U-235  Y 1.84E-07 U-238   Y 1.45E-06 NP-237  W 3.54E-11 PU-238  Y 1.39E-04
 PU-239 Y 1.29E-04 PU-241  Y 5.61E-03 PU-242  Y 2.82E-07 AM-241  W 9.22E-05
 AM-243 W 6.23E-06 CM-243  W 6.38E-08 CM-244  W 6.08E-05

P-DERESIN - REGULAR WASTE STREAM
 H-3    * 1.11E+02 C-14    * 4.06E+00 FE-55   Y 9.75E+01 CO-60   Y 1.89E+02
 NI-59  W 1.16E-01 NI-63   W 3.59E+01 SR-90   D 8.08E+00 NB-94   Y 3.68E-03
 TC-99  D 3.43E-02 I-129   D 1.02E-01 CS-135  D 3.43E-02 CS-137  D 9.13E+02
 U-235  Y 1.96E-03 U-238   Y 1.55E-02 NP-237  W 3.78E-07 PU-238  Y 1.08E+00
 PU-239 Y 7.58E-01 PU-241  Y 3.31E+01 PU-242  Y 1.66E-03 AM-241  W 7.79E-01
 AM-243 W 5.25E-02 CM-243  W 4.13E-04 CM-244  W 3.75E-01

P-DEFILCR - REGULAR WASTE STREAM
 H-3    * 3.46E-01 C-14    * 1.28E-02 FE-55   Y 1.67E+02 CO-60   Y 3.22E+02
 NI-59  W 1.99E-01 NI-63   W 6.14E+01 SR-90   D 2.53E-02 NB-94   Y 6.29E-03
 TC-99  D 1.08E-04 I-129   D 3.19E-04 CS-135  D 1.08E-04 CS-137  D 2.87E+00
 U-235  Y 1.10E-04 U-238   Y 8.64E-04 NP-237  W 2.11E-08 PU-238  Y 7.56E-02
 PU-239 Y 1.14E-01 PU-241  Y 5.00E+00 PU-242  Y 2.51E-04 AM-241  W 4.94E-02
 AM-243 W 3.31E-03 CM-243  W 5.81E-05 CM-244  W 3.31E-02

P-DEEVAPB - REGULAR WASTE STREAM
 H-3    * 2.31E-01 C-14    * 1.46E-02 FE-55   Y 2.85E+01 CO-60   Y 4.76E+01
 NI-59  W 2.94E-02 NI-63   W 6.45E-01 SR-90   D 4.42E-02 NB-94   Y 9.29E-04
 TC-99  D 9.38E-04 I-129   D 2.49E-03 CS-135  D 9.38E-04 CS-137  D 2.49E+01
 U-235  Y 1.29E-05 U-238   Y 1.02E-04 NP-237  W 2.48E-09 PU-238  Y 7.46E-02
 PU-239 Y 3.54E-02 PU-241  Y 1.73E+00 PU-242  Y 7.36E-05 AM-241  W 4.50E-02
 AM-243 W 3.03E-03 CM-243  W 9.71E-05 CM-244  W 7.69E-02

B-DECORES - ACTIVATED METAL WASTE STREAM
 H-3    * 1.73E+01 C-14    * 9.71E+00 CL-36   W 2.13E-01 FE-55   Y 7.85E+04
 CO-60  Y 4.26E+04 NI-59   W 5.72E+01 NI-63   W 7.58E+03 SR-90   D 2.00E-01
 NB-94  Y 8.40E-02 TC-99   D 2.00E-02 I-129   D 5.59E-08 CS-135  D 3.59E-06
 CS-137 D 2.00E-01 U-233   Y 2.13E-04 PU-239  Y 3.59E-03            1.26E+03

B-DEACINT - ACTIVATED METAL WASTE STREAM
 H-3    * 4.03E-01 C-14    * 2.26E-01 CL-36   W 4.96E-03 FE-55   Y 1.83E+03
 CO-60  Y 9.91E+02 NI-59   W 1.33E+00 NI-63   W 1.77E+02 SR-90   D 4.65E-03
 NB-94  Y 1.95E-03 TC-99   D 4.65E-04 I-129   D 1.30E-09 CS-135  D 8.36E-08
 CS-137 D 4.65E-03 U-233   Y 4.96E-06 PU-239  Y 8.36E-05            2.93E+01
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

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B-DEACVES - ACTIVATED METAL WASTE STREAM
 H-3    * 9.80E-02 C-14   * 1.91E-02 CL-36  W 3.91E-04 FE-55  Y 1.60E+02
 CO-60  Y 9.39E+01 NI-59  W 1.25E-01 NI-63  W 1.56E+01 SR-90  D 9.80E-05
 NB-94  Y 1.91E-04 TC-99  D 5.86E-05 I-129  D 2.97E-11 CS-135 D 2.50E-09
 CS-137 D 1.37E-04 U-233  Y 8.63E-07 PU-239 Y 2.27E-05            1.07E-03

B-DEACTCO - REGULAR WASTE STREAM
 H-3    * 5.86E-01 C-14   * 2.12E-04 CL-36  W 1.09E-05 FE-55  Y 1.26E+00
 CO-60  Y 5.19E-02 NI-59  W 7.66E-06 NI-63  W 9.53E-04 SR-90  D 5.23E-06
 NB-94  Y 6.53E-07 TC-99  D 2.34E-08 I-129  D 1.63E-12 CS-135 D 1.06E-10
 CS-137 D 5.66E-06 U-233  Y 1.04E-07 PU-239 Y 9.19E-07            2.76E-03

B-DECONME - REGULAR WASTE STREAM
 H-3    * 3.93E-05 C-14   * 1.87E-06 FE-55  Y 1.48E-01 CO-60  Y 2.66E-01
 NI-59  W 1.64E-04 NI-63  W 1.35E-02 SR-90  D 1.56E-04 NB-94  Y 5.19E-06
 TC-99  D 4.37E-08 I-129  D 1.22E-07 CS-135 D 4.37E-08 CS-137 D 1.37E-03
 U-235  Y 2.49E-07 U-238  Y 1.97E-06 NP-237 W 4.80E-11 PU-238 Y 4.89E-03
 PU-239 Y 6.44E-03 PU-241 Y 1.29E-01 PU-242 Y 1.41E-05 AM-241 W 1.93E-05
 AM-243 W 1.31E-06 CM-243 W 1.26E-06 CM-244 W 1.20E-05

B-DECONCO - REGULAR WASTE STREAM
 H-3    * 3.04E-06 C-14   * 1.45E-07 FE-55  Y 1.14E-02 CO-60  Y 2.05E-02
 NI-59  W 1.27E-05 NI-63  W 1.04E-03 SR-90  D 1.21E-05 NB-94  Y 4.01E-07
 TC-99  D 3.37E-09 I-129  D 9.41E-09 CS-135 D 3.37E-09 CS-137 D 1.06E-04
 U-235  Y 1.93E-08 U-238  Y 1.52E-07 NP-237 W 3.72E-12 PU-238 Y 3.77E-04
 PU-239 Y 4.97E-04 PU-241 Y 9.98E-03 PU-242 Y 1.09E-06 AM-241 W 1.49E-06
 AM-243 W 1.01E-07 CM-243 W 9.77E-08 CM-244 W 9.24E-07

B-DETRASH - REGULAR WASTE STREAM
 H-3    * 1.52E-03 C-14   * 9.40E-05 FE-55  Y 1.35E-01 CO-60  Y 2.27E-01
 NI-59  W 1.40E-04 NI-63  W 3.06E-03 SR-90  D 2.86E-04 NB-94  Y 4.42E-06
 TC-99  D 6.03E-06 I-129  D 1.61E-05 CS-135 D 6.03E-06 CS-137 D 1.61E-01
 U-235  Y 2.75E-08 U-238  Y 2.16E-07 NP-237 W 5.29E-12 PU-238 Y 5.16E-05
 PU-239 Y 2.61E-05 PU-241 Y 1.27E-03 PU-242 Y 5.68E-08 AM-241 W 2.18E-05
 AM-243 W 1.47E-06 CM-243 W 4.35E-08 CM-244 W 3.39E-04

B-DERESIN - REGULAR WASTE STREAM
 H-3    * 2.24E-02 C-14   * 1.39E-03 FE-55  Y 1.10E+00 CO-60  Y 1.86E+00
 NI-59  W 1.14E-03 NI-63  W 2.51E-02 SR-90  D 4.25E-03 NB-94  Y 3.61E-05
 TC-99  D 8.91E-05 I-129  D 2.38E-04 CS-135 D 8.91E-05 CS-137 D 2.38E+00
 U-235  Y 6.21E-08 U-238  Y 4.91E-07 NP-237 W 1.19E-11 PU-238 Y 9.72E-05
 PU-239 Y 6.21E-05 PU-241 Y 3.04E-03 PU-242 Y 1.37E-07 AM-241 W 2.71E-05
 AM-243 W 1.83E-06 CM-243 W 3.15E-08 CM-244 W 2.12E-05

B-DEEVAPB - REGULAR WASTE STREAM
 H-3    * 1.69E-01 C-14   * 1.05E-02 FE-55  Y 2.05E+01 CO-60  Y 3.44E+01
 NI-59  W 2.12E-02 NI-63  W 4.66E-01 SR-90  D 3.20E-04 NB-94  Y 6.71E-04
 TC-99  D 6.77E-04 I-129  D 1.80E-03 CS-135 D 6.77E-04 CS-137 D 1.80E+01
 U-235  Y 9.28E-06 U-238  Y 7.34E-05 NP-237 W 1.79E-09 PU-238 Y 5.38E-02
 PU-239 Y 2.55E-02 PU-241 Y 1.24E+00 PU-242 Y 5.58E-05 AM-241 W 3.25E-02
 AM-243 W 2.20E-03 CM-243 W 7.01E-05 CM-244 W 5.55E-02
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

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W-THORHLW - REGULAR WASTE STREAM
 CO-60  Y 3.19E+01 SR-90  D 1.38E+04 CS-134 D 1.15E+01 CS-137 D 1.47E+04
 EU-154 W 8.94E+01 U-235  Y 4.00E-05 U-238  Y 4.10E-04 NP-237 W 1.15E-02

W-PUREHLW - REGULAR WASTE STREAM
 H-3    * 1.20E+00 SR-90  D 3.35E+03 TC-99  D 9.50E-01 RU-106 Y 5.50E-02
 SB-125 W 3.05E+00 SN-126 W 2.00E-02 I-129  D 2.35E-03 CS-134 D 1.05E+01
 CS-135 D 1.75E-02 CS-137 D 4.45E+03 EU-152 W 2.05E-01 EU-154 W 6.50E+01
 NP-237 W 4.48E-10 PU-238 Y 7.50E-01 PU-239 Y 1.39E+00 PU-241 Y 3.50E+01
 PU-242 Y 5.00E-04 AM-241 W 1.00E+01 AM-243 W 1.10E-01 CM-242 W 5.00E-04
 CM-244 W 4.40E+00          7.68E+03

W-COTRASH - REGULAR WASTE STREAM
 H-3    * 4.80E-12 C-14   * 5.44E-12 NI-63  W 3.84E-11 SR-90  D 3.07E-04
 TC-99  D 3.20E-08 RU-106 Y 1.41E-07 SB-125 W 6.08E-07 SN-126 W 1.60E-09
 I-129  D 8.32E-15 CS-134 D 2.66E-07 CS-135 D 4.48E-09 CS-137 D 3.20E-04
 EU-152 W 2.11E-08 EU-154 W 8.00E-06 NP-237 W 1.96E-09 PU-238 Y 3.52E-07
 PU-239 Y 1.12E-07 PU-241 Y 4.48E-06 PU-242 Y 6.72E-11 AM-241 W 4.80E-07
 AM-243 W 9.28E-09 CM-242 W 8.32E-09 CM-243 W 1.34E-09 CM-244 W 4.16E-07

W-NCSOLID - REGULAR WASTE STREAM
 H-3    * 2.10E-11 C-14   * 2.38E-11 NI-63  W 1.68E-10 SR-90  D 1.34E-03
 TC-99  D 1.40E-07 RU-106 Y 6.16E-07 SB-125 W 2.66E-06 SN-126 W 7.00E-09
 I-129  D 3.64E-14 CS-134 D 1.16E-06 CS-135 D 1.96E-08 CS-137 D 1.40E-03
 EU-152 W 9.24E-08 EU-154 W 3.50E-05 NP-237 W 1.96E-09 PU-238 Y 1.54E-06
 PU-239 Y 4.90E-07 PU-241 Y 1.96E-05 PU-242 Y 2.94E-10 AM-241 W 2.10E-06
 AM-243 W 4.06E-08 CM-242 W 3.64E-08 CM-243 W 5.88E-09 CM-244 W 1.82E-06

W-LLWTFRE - REGULAR WASTE STREAM
 H-3    * 1.11E-10 C-14   * 1.25E-10 NI-63  W 8.84E-10 SR-90  D 7.07E-03
 TC-99  D 7.37E-07 RU-106 Y 3.24E-06 SB-125 W 1.40E-05 SN-126 W 3.68E-08
 I-129  D 1.92E-13 CS-134 D 6.11E-06 CS-135 D 1.03E-07 CS-137 D 7.37E-03
 EU-152 W 4.86E-07 EU-154 W 1.84E-04 NP-237 W 1.03E-08 PU-238 Y 8.10E-06
 PU-239 Y 2.58E-06 PU-241 Y 1.03E-04 PU-242 Y 1.55E-09 AM-241 W 1.11E-05
 AM-243 W 2.14E-07 CM-242 W 1.92E-07 CM-243 W 3.09E-08 CM-244 W 9.58E-06

W-FRSRESN - REGULAR WASTE STREAM
 SR-90  D 1.86E+00 CS-137 D 1.98E+01

W-FRSLIQD - REGULAR WASTE STREAM
 SR-90  D 3.80E-01 CS-137 D 3.36E+00

W-RTSRESN - REGULAR WASTE STREAM
 SR-90  D 3.28E-01 CS-137 D 2.95E+00
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

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W-LTTRASH - REGULAR WASTE STREAM
 H-3    * 2.14E-10 C-14   * 2.43E-10 NI-63  W 1.71E-09 SR-90  D 1.37E-02
 TC-99  D 1.43E-06 RU-106 Y 6.28E-06 SB-125 W 2.17E-05 SN-126 W 7.14E-08
 I-129  D 3.71E-13 CS-134 D 1.19E-05 CS-135 D 2.00E-07 CS-137 D 1.43E-02
 EU-152 W 9.42E-07 EU-154 W 3.57E-04 NP-237 W 2.00E-08 PU-238 Y 1.57E-05
 PU-239 Y 5.00E-06 PU-241 Y 2.00E-04 PU-242 Y 3.00E-09 AM-241 W 2.14E-05
 AM-243 W 4.14E-07 CM-242 W 3.71E-07 CM-243 W 6.00E-08 CM-244 W 1.86E-05

W-HTTRASH - REGULAR WASTE STREAM
 H-3    * 3.62E-10 C-14   * 4.10E-10 NI-63  W 2.89E-09 SR-90  D 2.32E-02
 TC-99  D 2.41E-06 RU-106 Y 1.06E-05 SB-125 W 4.58E-05 SN-126 W 1.21E-07
 I-129  D 6.27E-13 CS-134 D 2.00E-05 CS-135 D 3.38E-07 CS-137 D 2.41E-02
 EU-152 W 1.59E-06 EU-154 W 6.03E-04 NP-237 W 8.54E-06 PU-238 Y 6.71E-03
 PU-239 Y 2.13E-03 PU-241 Y 3.38E-04 PU-242 Y 1.28E-06 AM-241 W 9.15E-03
 AM-243 W 1.77E-04 CM-242 W 6.27E-07 CM-243 W 2.56E-05 CM-244 W 7.93E-03

W-LTEQUIP - REGULAR WASTE STREAM
 H-3    * 8.32E-08 C-14   * 9.43E-08 NI-63  W 6.66E-07 SR-90  D 5.33E+00
 TC-99  D 5.55E-04 RU-106 Y 2.44E-03 SB-125 W 1.05E-02 SN-126 W 2.77E-05
 I-129  D 1.44E-10 CS-134 D 4.61E-03 CS-135 D 7.77E-05 CS-137 D 5.55E+00
 EU-152 W 3.66E-04 EU-154 W 1.39E-01 NP-237 W 7.77E-06 PU-238 Y 6.10E-03
 PU-239 Y 1.94E-03 PU-241 Y 7.77E-02 PU-242 Y 1.17E-06 AM-241 W 8.32E-03
 AM-243 W 1.61E-04 CM-242 W 1.44E-04 CM-243 W 2.33E-05 CM-244 W 7.21E-03

W-HTEQUIP - REGULAR WASTE STREAM
 H-3    * 3.00E-07 C-14   * 3.41E-07 NI-63  W 2.40E-06 SR-90  D 1.92E+01
 TC-99  D 2.00E-03 RU-106 Y 8.81E-03 SB-125 W 3.81E-02 SN-126 W 1.00E-04
 I-129  D 5.21E-10 CS-134 D 1.66E-02 CS-135 D 2.80E-04 CS-137 D 2.00E+01
 EU-152 W 1.32E-03 EU-154 W 5.01E-01 NP-237 W 2.88E-04 PU-238 Y 2.26E-01
 PU-239 Y 7.20E-02 PU-241 Y 2.80E-01 PU-242 Y 4.32E-05 AM-241 W 3.08E-01
 AM-243 W 5.96E-03 CM-242 W 5.21E-04 CM-243 W 8.64E-04 CM-244 W 2.67E-01

W-PDWLIQD - REGULAR WASTE STREAM
 SR-90  D 2.45E+01 TC-99  D 2.55E-03 CS-137 D 2.55E+01 PU-238 Y 2.80E-02
 PU-239 Y 8.92E-03 PU-241 Y 3.57E-01 AM-241 W 3.82E-02

W-VITSUPR - REGULAR WASTE STREAM
 H-3    * 1.40E-04 C-14   * 1.40E-04 NI-63  W 5.60E-04 SR-90  D 3.60E-02
 TC-99  D 1.10E-01 RU-106 Y 2.40E-02 SN-126 W 2.80E-04 SB-125 W 1.10E-03
 CS-134 D 4.20E-04 CS-137 D 5.30E-01 EU-152 W 3.80E-02 EU-154 W 6.00E-02
 PU-238 Y 9.00E-03 PU-239 Y 3.00E-03 PU-241 Y 1.30E-01          9.01E-01

W-VITWASH - REGULAR WASTE STREAM
 SR-90  D 2.00E-01 TC-99  D 2.90E+00 RU-106 Y 1.30E+00 CS-137 D 2.90E+01
 EU-152 W 1.00E+00 EU-154 W 1.70E+00 PU-238 Y 2.40E-01 PU-241 Y 1.80E-01

W-VITSCRB - REGULAR WASTE STREAM
 SR-90  D 7.00E-01 CS-137 D 7.00E+01
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

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W-VITMELT - REGULAR WASTE STREAM
 SR-90  D 1.23E+03 CS-137 D 1.54E+03

W-VITFRAC - REGULAR WASTE STREAM
 SR-90  D 6.59E-04 CS-134 D 5.70E-07 CS-135 D 9.62E-09 CS-137 D 6.16E-02
 PU-238 Y 7.56E-07 PU-239 Y 2.40E-07 PU-241 Y 9.62E-06 AM-241 W 1.03E-06

W-VITZEOL - REGULAR WASTE STREAM
 SR-90  D 3.99E-03 CS-134 D 3.44E-06 CS-135 D 5.81E-08 CS-137 D 3.78E+00
 PU-238 Y 4.56E-06 PU-239 Y 1.45E-06 PU-241 Y 5.81E-05 AM-241 W 6.23E-06

W-DDRACKS - REGULAR WASTE STREAM
 H-3    * 1.02E-09 C-14   * 1.16E-09 NI-63  W 8.18E-09 SR-90  D 6.55E-02
 TC-99  D 6.82E-06 RU-106 Y 3.00E-05 SB-125 W 1.30E-04 SN-126 W 3.41E-07
 I-129  D 1.77E-12 CS-134 D 5.66E-05 CS-135 D 9.55E-07 CS-137 D 6.82E-02
 EU-152 W 4.50E-06 EU-154 W 1.71E-03 NP-237 W 9.55E-08 PU-238 Y 7.50E-05
 PU-239 Y 2.39E-05 PU-241 Y 9.55E-04 PU-242 Y 1.43E-08 AM-241 W 1.02E-04
 AM-243 W 1.98E-06 CM-242 W 1.77E-06 CM-243 W 2.86E-07 CM-244 W 8.87E-05

W-DDLTRUB - REGULAR WASTE STREAM
 H-3    * 1.36E-10 C-14   * 1.55E-10 NI-63  W 1.09E-09 SR-90  D 8.73E-03
 TC-99  D 9.10E-07 RU-106 Y 4.00E-06 SB-125 W 1.73E-05 SN-126 W 4.55E-08
 I-129  D 2.36E-13 CS-134 D 7.55E-06 CS-135 D 1.27E-07 CS-137 D 9.10E-03
 EU-152 W 6.00E-07 EU-154 W 2.27E-04 NP-237 W 1.27E-08 PU-238 Y 1.00E-05
 PU-239 Y 3.18E-06 PU-241 Y 1.27E-04 PU-242 Y 1.91E-09 AM-241 W 1.36E-05
 AM-243 W 2.64E-07 CM-242 W 2.36E-07 CM-243 W 3.82E-08 CM-244 W 1.18E-05

W-DDHTRUB - REGULAR WASTE STREAM
 H-3    * 1.53E-10 C-14   * 1.73E-10 NI-63  W 1.22E-09 SR-90  D 9.77E-03
 TC-99  D 1.02E-06 RU-106 Y 4.48E-06 SB-125 W 1.93E-05 SN-126 W 5.09E-08
 I-129  D 2.65E-13 CS-134 D 8.45E-06 CS-135 D 1.43E-07 CS-137 D 1.02E-02
 EU-152 W 6.72E-07 EU-154 W 2.55E-04 NP-237 W 7.32E-05 PU-238 Y 5.75E-02
 PU-239 Y 1.83E-02 PU-241 Y 1.43E-04 PU-242 Y 1.10E-05 AM-241 W 7.84E-02
 AM-243 W 1.52E-03 CM-242 W 2.65E-07 CM-243 W 2.19E-04 CM-244 W 6.79E-02

W-DDLTLQD - REGULAR WASTE STREAM
 H-3    * 1.05E-06 C-14   * 1.19E-06 NI-63  W 8.40E-06 SR-90  D 6.72E+01
 TC-99  D 7.00E-03 RU-106 Y 3.08E-02 SB-125 W 1.33E-01 SN-126 W 3.50E-04
 I-129  D 1.82E-09 CS-134 D 5.81E-02 CS-135 D 9.80E-04 CS-137 D 7.00E+01
 EU-152 W 4.62E-03 EU-154 W 1.75E-00 NP-237 W 9.80E-05 PU-238 Y 7.70E-02
 PU-239 Y 2.45E-02 PU-241 Y 9.80E-01 PU-242 Y 1.47E-05 AM-241 W 1.05E-01
 AM-243 W 2.03E-03 CM-242 W 1.82E-03 CM-243 W 2.94E-04 CM-244 W 9.10E-02

W-DDHTLQD - REGULAR WASTE STREAM
 H-3    * 7.38E-06 C-14   * 8.36E-06 NI-63  W 5.90E-05 SR-90  D 4.72E+02
 TC-99  D 4.92E-02 RU-106 Y 2.16E-01 SB-125 W 9.35E-01 SN-126 W 2.46E-03
 I-129  D 1.28E-08 CS-134 D 4.08E-01 CS-135 D 6.89E-03 CS-137 D 4.92E+02
 EU-152 W 3.25E-02 EU-154 W 1.23E+01 NP-237 W 1.47E-04 PU-238 Y 1.15E-01
 PU-239 Y 3.67E-02 PU-241 Y 6.89E+00 PU-242 Y 2.20E-05 AM-241 W 1.57E-01
 AM-243 W 3.04E-03 CM-242 W 1.28E-02 CM-243 W 4.14E-04 CM-244 W 1.36E-01
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TABLE A-75 . List of Waste Streams and Radionuclides (continued)

W-DDRESIN - REGULAR WASTE STREAM

H-3	*	2.94E-06	C-14	*	3.33E-06	NI-63	W	2.35E-05	SR-90	D	1.88E+02
TC-99	D	1.96E-02	RU-106	Y	8.62E-02	SB-125	W	3.72E-01	SN-126	W	9.80E-04
I-129	D	5.10E-09	CS-134	D	1.63E-01	CS-135	D	2.74E-03	CS-137	D	1.96E+02
EU-152	W	1.29E-02	EU-154	W	4.90E+00	NP-237	W	2.74E-04	PU-238	Y	2.16E-01
PU-239	Y	6.86E-02	PU-241	Y	2.74E+00	PU-242	Y	4.12E-05	AM-241	W	2.94E-01
AM-243	W	5.68E-03	CM-242	W	5.10E-03	CM-243	W	8.23E-04	CM-244	W	2.55E-01

L-SPENTFU - REGULAR WASTE STREAM

H-3	*	1.04E+03	C-14	*	1.98E+00	FE-55	Y	2.22E+02	CO-60	Y	2.47E+02
NI-59	W	7.41E-02	NI-63	W	9.88E+00	SR-90	D	1.61E+05	TC-99	D	3.21E+01
RU-106	Y	4.20E+05	CM-244	W	3.21E+03	SN-126	W	1.19E+00	I-129	D	8.15E-02
CS-135	D	6.67E-01	CS-137	D	2.27E+05	EU-152	W	2.96E+01	EU-154	W	1.36E+04
U-234	Y	5.19E-02	U-235	Y	3.95E-02	U-236	Y	5.43E-01	U-238	Y	7.90E-01
NP-237	W	7.66E-01	PU-236	Y	5.68E-01	PU-238	Y	5.19E+03	PU-239	Y	7.16E+02
PU-240	Y	1.11E+03	PU-241	Y	2.72E+05	PU-242	Y	3.95E+00	AM-241	W	9.14E+02
AM-243	W	3.46E+01	CM-242	W	8.89E+03	CM-243	W	9.63E+00			2.47E+06

L-FUEHARD - ACTIVATED METAL WASTE STREAM

C-14	*	8.90E-01	FE-55	Y	7.10E+04	CO-60	Y	7.10E+04	NI-59	W	5.40E+01
NI-63	W	7.40E+03	SR-90	D	1.80E-02	NB-94	Y	1.80E-03	TC-99	D	1.30E-01

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APPENDIX B

WASTE PROCESSING AND SPECTRA OPTIONS

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APPENDIX B. WASTE PROCESSING AND SPECTRA OPTIONS

B.1 INTRODUCTION

The projections of waste stream volumes and radionuclide concentrations made in the previous Appendix A were for "untreated" waste. Sometimes some processing had been performed in order to generate the waste stream, but in all cases the volumes and radionuclide concentrations were given prior to consideration of waste packaging. And in many cases, additional processing of many waste streams is routinely performed by licensees. Thus, these possible packaging and processing options must be considered for a complete analysis.

Other considerations include the fact that various optional waste treatment and packaging techniques will alter several physical and radiological characteristics that are important in determining impacts from radioactive waste management. For example, compaction of compressible, combustible trash will increase the radionuclide concentrations in the waste; incineration will further increase concentrations while the flammability of the waste will be reduced. Solidification of spent ion exchange resins prior to disposal will improve the long-term stability of the waste, albeit at a larger monetary cost than disposal of the resins in a dewatered form.

This appendix ties these considerations together. First, the various waste treatment and packaging options considered in this report are addressed. This information is mainly a summary of information previously supplied in the data base reports (Refs. 1, 2) and the draft Part 61 environmental impact statement (EIS) (Ref. 3), but is updated as necessary--particularly in terms of unit impact measures such as waste processing costs. Second, the optional "waste spectra" information is presented. This is merely a discussion of (1) the physical characteristics of the waste considered important for determining impacts, (2) the waste processing and packaging options actually considered for each waste stream, and (3) the manner in which the physical and radiological characteristics are manipulated in the computer codes (the waste spectral and other indices).

The waste spectral and other integers are used as qualitative rankings of particular waste properties, and also to trigger specific computational procedures in the impacts analysis computer codes. Many of the waste form behavior index values are assigned in a rather subjective way. However, this is not believed to be a major problem since the major purpose of the impact analysis methodology is to enable the user to compare, and quantify to a certain extent, trends associated with particular alternative rulemaking actions. The waste form behavior indices should not be used as a substitute for real data and test results for specific waste streams.

It should also be noted that three high-level liquid waste streams are included in the report, and that these waste streams are hypothetically assumed to be processed and solidified like any other liquid generated by the nuclear industry.

This is contrary to existing laws, regulations, national plans, and (probably) technical feasibility. However, handling these high level liquid waste streams in this hypothetical way allows comparison of potential impacts of waste processing and disposal over a wide variety of wastes. It serves to provide, for comparison purposes, an upper bound for impacts from near-surface waste disposal.

For the reader's reference, a list of all the waste streams considered in this report are included in Table B-1.

B.2 WASTE PROCESSING AND PACKAGING

There are many processing and packaging techniques currently available that can be used to alter the physical and radiological characteristics of radioactive waste forms. This section briefly considers several of these technologies and presents their estimated impacts on waste generators and/or waste processors. The discussion in this section is updated from the discussions previously presented in the data base reports and draft Part 61 EIS (Refs. 1-3).

The effects of the different waste processing and packaging options are compared through the use of three impact measures. These impact measures include occupational exposures, population exposures, and costs. Only incineration is assumed to result in effluents being released during waste processing which result in potentially significant population exposures. Other treatment processes, including evaporation, compaction, solidification, and packaging, are assumed to not result in significant additional potential impacts off of the processing facility site.

Waste processing and packaging options are considered in three sections. Chapter B.2.1 addresses treatment processes that result in a reduced volume after processing. Chapter B.2.2 addresses treatment processes that result in an increased volume after processing. Chapter B.2.3 addresses waste packaging options, including possible use of high integrity containers.

B.2.1 Volume Reduction

There are three basic processes that can be applied to waste streams that result in overall waste volume reduction: (1) physical processes such as compaction, (2) thermal processes such as evaporation, and (3) incineration and other related combustion processes. Each of these processes produces a concentrate stream and an effluent stream. The respective concentrate streams are compressed wastes, concentrated liquids or crystals, and ash. The respective effluents displaced are air, vapor, and gas and vapor. The activity per unit volume of the concentrate stream is usually higher than that of the untreated waste with the possible exception of volatile nuclides such as tritium, carbon, and iodine that may be entrained as vapor and/or combustion products in the effluent stream.

The volume reduction factor (VRF) is defined in this appendix as the ratio of the waste volume that is input to the process (untreated volume) to that of the concentrated (treated) waste volume.

Table B-1. Waste Groups and Streams

I. Nuclear Power Plants	Symbols
PWR ion-exchange resins	P-IXRESIN
PWR concentrated liquid	P-CONCLIQ
PWR filter sludges	P-FSLUDGE
PWR cartridge filters	P-FCARTRG
BWR ion-exchange resins	B-IXRESIN
BWR concentrated liquids	B-CONCLIQ
BWR filter sludges	B-FSLUDGE
PWR combustible trash	P-COTRASH
PWR noncombustible trash	P-NCTRASH
BWR combustible trash	B-COTRASH
BWR noncombustible trash	B-NCTRASH
LWR nonfuel reactor core components	L-NFRCOMP
LWR decontamination waste	L-DECONRS
II. Other Nuclear Fuel Cycle Facilities	
Fuel fabrication process waste	F-PROCESS
Fuel fabrication combustible trash	F-COTRASH
Fuel fabrication noncomb trash	F-NCTRASH
UF_6 conversion process waste	U-PROCESS
Pu facility decontamination waste	L-PUDECON
Waste from hot cell burnup studies	L-BURNUPS
III. Institutional Waste	
Combustible trash (LF)[a]	I-COTRASH
Combustible trash (SF)[b]	I+COTRASH
Absorbed liquids (LF)	I-ABSLIQD
Absorbed liquids (SF)	I+ABSLIQD
Liquid scintillation vial waste (LF)	I-LIQSCVL
Liquid scintillation vial waste (SF)	I+LIQSCVL
Biological waste (LF)	I-BIOWAST
Biological waste (SF)	I+BIOWAST
IV. Industrial Waste	
Source and SNM trash (LF)	N-SSTRASH
Source and SNM trash (SF)	N+SSTRASH
Source and SNM waste	N-SSWASTE
Low Activity trash (LF)	N-LOTRASH
Low activity trash (SF)	N+LOWTRASH
Low activity waste	N-LOWASTE

IV. Industrial Waste (cont'd)	Symbols
Large Radioisotope Manufacturers	
High-activity production trash	N-ISOPROD
Low-activity production trash	N-ISOTRSH
Large sealed source manufacturers	N-SORMFG1
	N-SORMFG2
	N-SORMFG3
	N-SORMFG4
Large Tritium and Carbon-14 Manfacturers	
Compactible trash	N-NECOTRA
Absorbed organic liquid	N-NEABLIQ
Solidified aqueous liquid	N-NESOLIQ
Reject product vials	N-NEVIALS
Noncompactible glass	N-NENCGLS
Noncompactible wood/metal	N-NEWOTAL
Tritium gas	N-NETRGAS
Absorbed tritiated liquid	N-NETRILI
Absorbed C-14 liquid	N-NECARLI
Laboratory trash	N-MWTRASH
Absorbed organic liquid	N-MWABLIQ
Solidified aqueous liquid	N-MWSOLIQ
Miscellaneous waste	N-MWWASTE
Small Tritium Manufacturers	
Tritium in paint or as plating	N-TRIPLAT
Gaseous tritium	N-TRITGAS
High-activity scintillation liquids	N-TRISCNT
Tritium in aqueous liquid	N-TRILIQD
Miscellanoues trash	N-TRITRSH
Tritium cont./absorbed in metal	N-TRIFOIL
High-activity waste	N-HIGHACT
Sealed sources and devices	
Tritium sources	N-TRITSOR
Carbon-14 sources	N-CARBSOR
Cobalt-60 sources	N-COBLSOR
Nickel-63 sources	N-NICKSOR
Strontium-90 sources	N-STROSOR
Cesium-137 sources	N-CESISOR
Plutonium-238 sources	N-PLU8SOR
Plutonium-239 sources	N-PLU9SOR
Americium-241 sources	N-AMERSOR
Pu-238 neutron sources	M-PUBESOR
Am-241 neutron sources	M-AMBESOR

Table B-1. Waste Groups and Streams (continued)

V. Other Non-Fuel Cycle Waste	Symbols
Radium sources	
Medical needles	N-RANEEDS
Medical cells	N-RACELLS
Medical plaques	N-RAPLAQU
Medical nasopharyngeal applicators	N-RANPAPP
Radium-beryllium neutron sources	N-RABESOR
Miscellaneous non-medical sources	N-RAMISCL
Radium ion-exchange resins	N-RARESIN
Navy wet waste	M-NAVYWET
Navy dry waste	M-NAVYDRY
VI. Non-Routine Waste	
Uranium Fuel Reprocessing	
High-level liquid waste	R-HLLWFRP
Fuel assembly hardware	R-FUEHARD
Hulls fron chop/leach process	R-HULLFRP
Intermedia-level liquid waste	R-ILLWFRP
Silica gel	R-SILIGEL
Main plant high-activity comp trash	R-MPCOTRH
Main plant low-activity comp trash	R-MPCOTRL
Main plant noncompressible trash	R-MPNCTRA
Degraded extractant	R-DEGREXT
Main plant ion-exchange resins	R-MPRESIN
Storage basin resin and filter sludge	R-SBRESIN
Storage basin concentrated liquids	R-SBCOLIQ
Storage basin compressible trash	R-SBCOTRA
Storage basin noncompressible trash	R-SBNCTRA
UF_6 conv. flourinator residues	R-UFFINES
UF_6 conv. K_2UO_4 mud	R-UFKZMUD
UF_6 conv. compressible trash	R-UFCOTRA
UF_6 conv. noncompressible trash	R-UFNCTRA
PuO_2 conv. compressible trash	R-PUCOTRA
PuO_2 conv. noncompressible trash	R-PUNCTRA
Mixed Oxide Fuel Fabrication	
Compressible trash	R-MOXCOTR
Noncompressible trash	R-MOXNCTR
Proc & scrap recovery solutions	R-MOXSOLN

VI. Non-Routine Waste (Cont'd)	Symbols
Nuclear Power Plant Decommissioning	
PWR activated core shroud	P-DECORES
PWR activitated reactor internals	P-DEACINT
PWR activated reactor vessel	P-DEACVES
PWR activated concrete	P-DEACTCO
PWR contaminated metals	P-DECONME
PWR contaminated concrete	P-DECONCO
PWR combustible/compactible trash	P-DETRASH
PWR chelated ion-exchanger resins	P-DERESIN
PWR filter cartridges	P-DEFILCR
PWR evaporator bottoms	P-DEEVAPB
BWR activated core shroud	B-DECORES
BWR activated reactor internals	B-DEACINT
BWR activated reactor vessel	B-DEACVES
BWR activated concrete	B-DEACTCO
BWR contaminated metals	B-DECONME
BWR contaminated concrete	B-DECONCO
BWR combustible/compactible trash	B-DETRASH
BWR chelated ion-exchange resins	B-DERESIN
BWR evaporator bottoms	B-DEEVAPB
West Valley Demonstration Project	
Thorex high-level waste	W-THORHLW
Purex high-level waste	W-PUREHLW
Trash from existing systems	W-COTRASH
Miscellaneous dry solids	W-NCSOLID
LLWTF sludge and resins	W-LLWTFRE
FRS filter precoat and resins	W-FRSRESN
RTS liquid waste	W-FRSLIQD
RTS filter backwash and resins	W-RTSRESN
Trash, low TRU content	W-LTTRASH
Trash, high TRU content	W-HTTRASH
Equipment and hardware, low TRU	W-LTEQUIP
Equipment and hardware, high TRU	W-HTEQUIP
PD liquid waste	W-PDWLIQD
Vitrification Waste	
Supernate	W-VITSUPR

Table B-1. Waste Groups and Streams (continued)

VI. Non-Routine Waste (Cont'd)	Symbols
Sludge wash	W-VITWASH
Scrub condensate	W-VITSCRB
Melter feed overhead	W-VITMELT
Fractionator condensate	W-VITFRAC
Zeolite slurry	W-VITZEOL
Decontamination & Decommission	
Fuel storage racks	W-DDRACKS
Low TRU rubble	W-DDLTRUB
High TRU rubble	W-DDHTRUB
Low TRU liquids	W-DDLTLQD
High TRU liquids	W-DDHTLQD
Resins	W-DDRESIN
VII. Other Waste	
Spent nuclear power plant fuel rods	L-SPENTFU
Fuel assembly hardware	L-FUEHARD

Note: a - Large Facility; b - Small Facility; c - Scintillation Liquids

Compaction

Compaction is an often-used method--particularly at nuclear fuel cycle facilities--of reducing the volume of waste streams containing compressible material such as paper, plastic, glass, wood, and light-gauge metal. Most of the volume reduction is attained by compressing the waste to reduce its void volume. The term compactor is usually applied to hydraulic or mechanical rams that compress wastes into boxes or 55-gallon steel drums. The boxes and drums are then used as disposal containers. Typical hydraulic rams generate 20,000 to 30,000 pounds of force, and are fitted with shrouds and simple air filtration systems to minimize release of airborne radioactivity.

Most compactors now in use can achieve average volume reduction factors of about 2. Newer compactors, which place a metal inner sleeve inside the drum during compaction, are capable of a volume reduction factor of about 4. Units which compact waste into large boxes are also being used in many facilities. Industrial hydraulic presses similar to those used to crush automobiles may be useful for compacting heavier gauge metal items such as pipes, tools, cans, drums, and scaffolding.

In this appendix, three types of compactors are considered: compactors that can be utilized to achieve volume reduction factors of around 1.5 to 3; improved compactor/shredders that can achieve volume reduction factors of about 6; and industrial hydraulic presses that are also assumed to be capable of achieving volume reduction factors of about 6. In the analysis, the improved compactor/shredders are assumed to be operable as an option by any facility capable of implementing its own processing system (fuel cycle facilities and large institutional and industrial facilities). Industrial hydraulic presses, however, are assumed to be operable only at a centralized waste processing facility.

The waste streams to which these compaction techniques are applied, and their unit impact measures, are summarized in Table B-2. Different volume reduction factors are applied to various compactors depending upon the waste form and type of facility generating the waste. This is especially true for the reference compaction option. The volume reduction factors for PWR and BWR compressible trash waste streams represent industry-wide averages and were estimated based on data obtained from reference 4.

Evaporation

Evaporators concentrate liquid wastes by heating them to vaporize the volatile components. The vaporized water generally contains greatly reduced quantities of dissolved solids, suspended solids, and radioactivity relative to those found in the input waste stream. In the nuclear industry the vaporized water is normally condensed and collected, and then either discharged or recycled after testing to determine whether the condensate requires additional treatment. The concentrated solution (bottoms) left in the evaporator retains virtually all of the solids and radioactivity and is solidified and shipped to a disposal facility.

Table B-2. Compaction Techniques and Impacts

Compaction Technique	Cost* per m^3	Man-Hours* per m^3	Waste Streams	Volume Reduction Factor	Waste Spectrum Considered
Compactor	$412	15	P-COTRASH	3.0	1, 2, 5
			B-COTRASH	2.0	1, 2, 5
			F-COTRASH	1.5	1, 2, 5
			I-COTRASH	2.0	1, 2, 5
			N-SSTRASH	1.5	1, 2, 5
			N-LOTRASH	2.0	1, 2, 5
			I-LQSCNVL	1.28	1, 2, 5
			N-NECOTRA	2.0	1, 2, 5
			R-MPCOTRH	3.0	1, 2, 5
			R-MPCOTRL	3.0	1, 2, 5
			R-SBCOTRA	3.0	1, 2, 5
			R-UFCOTRA	3.0	1, 2, 5
			R-PUCOTRA	3.0	1, 2, 5
			R-MOXCOTR	3.0	1, 2, 5
			P-DETRASH	3.0	1, 2, 5
			B-DETRASH	3.0	1, 2, 5
			W-COTRASH	3.0	1, 2, 5
			W-LTTRASH	3.0	1, 2, 5
			W-HTTRASH	3.0	1, 2, 5
Improved compactor/ shredder	$619	15	P-COTRASH	6.0	3
			B-COTRASH	6.0	3
			F-COTRASH	6.0	3
			I-COTRASH	6.0	3
			I+COTRASH	6.0	3
			N-SSTRASH	5.0	3
			N+SSTRASH	5.0	3
			N-LOTRASH	6.0	3
			N+LOTRASH	6.0	3
			N-NECOTRA	6.0	3
			R-MPCOTRH	6.0	3
			R-MPCOTRL	6.0	3
			R-SBCOTRA	6.0	3
			R-UFCOTRA	6.0	3
			R-PUCOTRA	6.0	3
			R-MOXCOTR	6.0	3
			P-DETRASH	6.0	3
			B-DETRASH	6.0	3
			W-COTRASH	6.0	3
			W-LTTRASH	6.0	3
			W-HTTRASH	6.0	3

Table B-2. (Continued)

Compaction Technique	Cost* per m^3	Man-Hours* per m^3	Waste Streams	Volume Reduction Factor	Waste Spectrum Considered
Industrial hydraulic press	$1237	15	P-NCTRASH	6.0	4
			B-NCTRASH	6.0	4
			F-NCTRASH	6.0	4
			N-ISOTRSH	6.0	4
			N-SORMFG1	6.0	4
			N-SORMFG4	6.0	4
			N-TRIPLAT	6.0	4
			N-TRITRSH	6.0	4
			N-TRIFOIL	6.0	4
			M-NAVYDRY	6.0	4
			R-MPNCTRA	6.0	4
			R-SBNCTRA	6.0	4
			R-UFNCTRA	6.0	4
			R-PUNCTRA	6.0	4
			R-MOXNCTR	6.0	4
			W-NCSOLID	6.0	4

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*Cost and man-hours are given in units of input (untreated) waste volume. Costs were estimated by multiplying the cost assumptions given in the data base report by a factor 1.23, which is the ratio of the producer price indices for capital equipment for the years 1980 and 1984 (Ref. 5).

Evaporators can be categorized according to their methods of heat transfer. Natural circulation evaporators use convection as the means of heat transfer.

Forced circulation evaporators use pumps to improve the flow of liquid over the heating surfaces. Fluidized-bed dryers produce dry salts by injecting atomized waste liquids onto a hot bed of inert granules that is suspended (fluidized) in a stream of hot air. The liquids flash-evaporate on contact with the hot bed, leaving behind a residue of dry solids. The inert carrier process uses a hot bath of inert fluid recirculating at high velocities as the heat exchanger. Solidification in bitumen can also be considered as a form of evaporation. The ideal evaporator produces a condensate that is free of radioactivity while attaining the maximum volume reduction of the input waste liquid.

In this appendix, evaporator/crystallizers are assumed to be utilized as an option to further concentrate the already concentrated liquid waste streams of LWRs. For the reference representative evaporator/crystallizer, the volume reduction factors assumed in this appendix are listed in Table B-3 as a function of waste stream. The impact measures are $849 and 4.42 man-hours per m^3 of (untreated) input waste liquid.

Table B-3. Use of Evaporator/Crystallizers

Waste Streams	Volume Reduction Factor	Waste Spectrum Considered
P-CONCLIQ	6.0	3
B-CONCLIQ	2.4	3
R-ILLWFRP	6.0	3
R-SBCOLIQ	6.0	3
R-MOXSOLN	6.0	3
P-DEEVAPB	6.0	3
B-DEEVAPB	2.4	3
W-FRSLIQD	6.0	3
W-PDWLIQD	6.0	3
W-VITSUPR	6.0	3
W-VITWASH	6.0	3
W-VITSCRB	6.0	3
W-VITMELT	6.0	3
W-VITZEOL	6.0	3
W-DDLTLQD	6.0	3
W-DDHTLQD	6.0	3

Incineration

Incinerators and related devices decompose combustible waste materials by thermal oxidation. Combustion or incineration involves complete oxidation of wastes by burning in an excess of oxygen (air). Pyrolysis involves partial oxidation in an oxygen-deficient atmosphere. Oxidation can also be accomplished by introducing combustible wastes and air into a bath of molten salt. Alternatively, acid digesters oxidize wastes in a hot mixture of concentrated nitric and sulfuric acids.

The various types of incinerators, pyrolyzers, and other such devices currently used or being developed for volume reduction of radioactive waste are too numerous to be considered here individually. Two reference types of representative incinerators have been selected for discussion in this appendix: pathological incinerators and fluidized bed incinerators. The reference pathological incinerator is considered for optional use by large institutional waste generators such as hospitals and biomedical research facilities. The reference fluidized bed incinerator is considered for optional use by fuel cycle waste generators or by operators of a potential regional waste processing facility incinerating wastes from small waste generators. The waste streams treated with these two types of incinerators and the resultant unit impact measures are presented in Table B-4.

Pathological incinerators are typically multiple-chamber, hot refractory hearth incinerators and are normally operated with less sophisticated off-gas treatment systems. Airborne releases are principally controlled through control of the rate of input feed. They are designed primarily for the incineration of animal carcasses and operate at approximately 900 to 1000°C. Pathological incinerators may also be used by institutional waste generators

Table B-4. Incineration Techniques and Impacts

Incineration Technique	Cost* per m^3	Man-Hours* per m^3	Waste Streams	Volume Reduction Factor	Waste Spectrum Considered
Pathological incinerator	$2534	8	I-COTRASH	20.0	4
			N-SSTRASH	10.0	4
			N-LOTRASH	20.0	4
			I-LQSCNVL	4.52	4
			I-ABSLIQD	100.0	4
			I-BIOWAST	15.0	4
			N-NECOTRA	20.0	4
			N-TRISCNT	4.52	4
			N-TRILIQD	100.0	4
Fluidized bed incinerator (at facilities)	$2384	6.12	P-IXRESIN	18.0	4
			P-CONCLIQ	8.0	4
			P-FSLUDGE	5.0	4
			B-IXRESIN	18.0	4
			B-CONCLIQ	6.4	4
			B-FSLUDGE	5.0	4
			P-COTRASH	80.0	4
			B-COTRASH	80.0	4
			F-COTRASH	40.0	4
			M-NAVYWET	8.0	4
			R-ILLWFRP	8.0	4
			R-MPCOTRH	80.0	4
			R-MPCOTRL	80.0	4
			R-DEGREXT	4.5	4
			R-MPRESIN	18.0	4
			R-SBRESIN	18.0	4
			R-SBCOLIQ	8.0	4
			R-SBCOTRA	80.0	4
			R-UFCOTRA	80.0	4
			R-PUCOTRA	80.0	4
			R-MOXCOTR	80.0	4
			R-MOXSOLN	8.0	4
			P-DETRASH	80.0	4
			P-DERESIN	18.0	4
			P-DEEVAPB	8.0	4
			B-DETRASH	80.0	4
			B-DERESIN	18.0	4
			B-DEEVAPB	6.4	4
			W-COTRASH	80.0	4
			W-LLWTFRE	5.0	4
			W-FRSRESN	5.0	4
			W-FRSLIQD	8.0	4
			W-RTSRESN	5.0	4
			W-LTTRASH	80.0	4
			W-HTTRASH	80.0	4
			W-PDWLIQD	8.0	4

Table B-4. (Continued)

Incineration Technique	Cost* per m^3	Man-Hours* per m^3	Waste Streams	Volume Reduction Factor	Waste Spectrum Considered
Fluidized bed incinerator (at facilities) (continued)	$2384	6.12	W-VITSUPR	8.0	4
			W-VITWASH	8.0	4
			W-VITSCRB	8.0	4
			W-VITMELT	8.0	4
			W-VITFRAC	8.0	4
			W-VITZEOL	8.0	4
			W-DDLTLQD	8.0	4
			W-DDHTLQD	8.0	4
			W-DDRESIN	18.0	4
Fluidized bed incinerator (at regional processing center)	$1278	5.35	I+COTRASH	80.0	4
			N+SSTRASH	40.0	4
			N+LOTRASH	80.0	4
			N-RARESIN	18.0	4

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*Cost and man-hours are given in unit volume of input (untreated) waste. Costs were estimated by multiplying the cost assumptions given in the data base report by a factor 1.23, which is the ratio of the producer price indices for capital equipment for the years 1980 and 1984 (Ref. 5).

for volume reduction of other biowastes, scintillation fluids, organic liquids, and trash. Aqueous liquids can also be evaporated on the refractory hearth.

Fluidized bed incinerators operate by injecting combustible wastes into a hot bed of inert granules fluidized by a stream of hot gas. They operate on the same principle as fluidized bed dryers or calciners which have been used for many years in nonnuclear industries to produce dry solids from liquid wastes by complete evaporation of the water. Typical fluidized bed incinerators can burn trash, organic solvents, and ion exchange resins. Wastes are normally screened to remove metal objects and shredded before entering the process vessel. The process vessel is maintained at 800 to 1000°C. Residual ash from the combustion process is collected for solidification. Ash carried out of the process vessel with the hot effluent gas stream is separated from the effluent gas by an off-gas treatment system, and also collected for subsequent solidification.

Thermal combustion is also apparently the most effective way of removing chelating agents and organic chemicals from waste streams.

B.2.2 Volume Increase

There are two basic processes that can be applied to waste streams that definitely result in an overall waste volume increase: solidification and addition of absorbent materials. The activity per unit volume of the product stream is generally lower than that of the input waste.

The volume increase factor (VIF) is defined in this appendix as the ratio of the volume of the treated waste product to the volume of the input untreated waste.

Solidification

This section considers a number of solidification processes that can be applied to waste streams such as LWR process wastes (concentrated liquids, resins, filter sludges, and cartridge filters), or dry salts and ashes produced by calciners and incinerators. Cartridge filters are assumed to be solidified by pouring the solidification agent into spaces between the currently utilized shipping containers and the cartridges. This results in no change to the currently shipped volume of the waste stream. Solidification of sealed sources is also considered as a special case.

The solidification agents or techniques considered in this appendix are selected from those that are currently in use or are being actively marketed. These include cement and synthetic polymer systems. Absorbents such as vermiculite and diatomaceous earth are not considered to be solidification agents since they do not chemically or physically bind the wastes. Cement and synthetic polymer solidification systems are currently commonly used by LWRs. Bitumen (another agent) is being actively marketed and some bitumen solidification systems (which are widely used in Europe) have been sold in this country. Polyester (another synthetic polymer) has been evaluated in laboratory and pilot plant studies using simulated LWR liquid wastes and may be routinely used in the future. Urea-formaldehyde was commonly used in the past, but has been prohibited from use at existing disposal sites given its inability to comply with the waste form requirements of 10 CFR Section 61.56.

In the analyses to determine the performance and technical requirements for disposal of LLW, three general solidification scenarios are postulated. These solidification scenarios are the same as those used for the analyses in reference 6, and are included to enable the code user to assess what level of overall impact reduction can be achieved given varying assumptions on solidified waste form leachability.

- o Solidification Scenario A assumes waste performance characteristics that are comparatively less desirable than the following two solidification scenarios. This is simulated by assuming that the waste stream is solidified using cement systems.

- o Solidification Scenario B assumes improved waste performance characteristics over the previous case. This is simulated by assuming that the waste stream is solidified using synthetic polymer systems.

- o Solidification Scenario C assumes an optimistic level of waste performance characteristics. This is simulated by assuming that the waste stream is all solidified using further improved synthetic polymer systems.

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These solidification processes, volume increase factors, and the unit impact measures associated with the processes are summarized in Table B-5. The listed costs do not include costs for process control programs which verify achievement of an acceptable stable waste form under 10 CFR 61.56, or properly classify waste. The volume increase factors were determined for the solidification scenarios by assuming volume increase factors of 1.4 for cement and 2.0 for synthetic polymer, and 2.0 for improved synthetic polymer.

Table B-5. Solidification Techniques and Impacts

Solidification Technique	Cost* per m^3	Man-Hours* per m^3
Scenario A	$1577	4
Scenario B	$3007	4
Scenario C	$3759	4

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*Cost and man-hours are given in unit volume of solidified (treated) waste. Costs were estimated by multiplying the cost assumptions given in reference 6 by a factor 1.23, which is the ratio of the producer price indices for the years 1980 and 1984 (Ref. 5).

Sealed sources are a special case. In this report an option is considered in which sources are stabilized by cementation within 55-gallon drums. For this option, costs are calculated by assuming that a two-person team is required: one to mix and pour cement, clean up afterwards, etc., and another to monitor the radiation environment during source emplacement and cementation. As a rough approximation:

labor:	4 hr x $10/hr =	40
	1 hr x $15/hr =	15
overhead:	x 30%	16.5
		71.5
cement:	$45/$m^3$ x .208m^3 =	9.4
		$80.9

Solidification costs per sealed source (less container) are therefore assumed to be roughly $80 per source. Radiation exposures are estimated by assuming 2 man-hours per source in a low background radiation environment.

Absorbent Materials

Absorbent materials such as diatomaceous earth or vermiculite are currently added to several institutional waste streams to minimize potential transportation impacts. These streams include liquid scintillation vial (LSV) waste, absorbed liquid waste, and biowaste. Existing commercial disposal facility operators require that these wastes be packaged with specified proportions of waste to absorbent material before they are accepted for disposal. For example, LSV waste is required to be packaged using sufficient absorbent material to absorb twice the total volume of the liquid in the package. Lime is frequently added to the biowaste stream. Double packaging of these waste streams is also used for additional safety. For the liquid scintillation vial and the absorbed liquid waste streams, a volume increase factor of 3.0 is assumed. For the biowaste stream, a volume increase factor of 1.92 is assumed.

The practice of packaging wastes with absorbent material increases the difficulty of processing these wastes with currently available methods, if delivered to a centralized processing facility. This is because many of the common absorbent materials, an integral part of the waste stream when the package leaves the waste generator, are not incinerable; absorbents that are incinerable are either not cost-effective or not compatible with the waste streams. Other processing techniques are either not compatible with the waste steams (e.g., cement solidification of liquid scintillation vials) or would result in an increase of the volume of the waste, and as a consequence would not be cost-effective. Therefore, these wastes would have to be processed by the waste generator. While many waste generators are capable of implementing their own waste processing alternatives, such as solidification instead of use of absorbent material, there is no alternative cost-effective treatment method (other than the use of absorbents) for small waste generators such as individual physicians, small medical groups, and small colleges for several waste streams. Therefore, the option of processing at a regional center was not implemented for the I+LQSCNVL, I+ABSLIQD, and I+BIOWAST waste streams.

B.2.3 Waste Packaging

Waste packaging may also result in an overall increase in waste volume where the complete container volume is not utilized. Generally the waste generator attempts to minimize void volume within containers. For purposes of determining the performance objectives and technical requirements for disposal, the waste volume increase due to packaging (which results in decreased radionuclide concentrations) is conservatively neglected. Moreover, there is little applicable data available on the packaging efficiency of waste streams. The uncertainties in other estimates in this appendix partially compensate for exclusion of packaging efficiency from volume calculations.

Generic Waste Containers Considered

Impact measures such as waste transportation costs, radiological exposures to transport vehicle loaders, and radiological exposures to disposal site workers are influenced by the types of waste containers used. To systematically quantify these impact measures, some estimate must be made of the types of containers used for each waste stream as a function of waste spectra. A very wide variety of waste containers is currently being used, and so some simplifying assumptions must be made.

Five generic types of waste containers are considered in this report: large boxes (128 ft^3), small boxes (16 ft^3), 55-gallon drums (7.5 ft^3), small liners (50 ft^3), and large liners (170 ft^3). Particular types of containers were assigned to particular waste streams based on input from a number of sources, including Appendix A, shipment records, and other references (Refs. 4, 7-11). The following general assignments are made:

Waste Streams	Containers
Institutional wastes	drums
Low activity industrial wastes	drums
Sealed sources	drums
Operating Reactor Waste	
LWR cartridge filters	drums
Other LWR process wastes	70% large liners, 15% small liners, 15% drums
PWR compressible trash	85% drums, 10% large boxes, 5% small boxes
BWR compressible trash	56% drums, 28% large boxes, 10% small boxes
LWR noncompressible trash	70% large boxes, 30% small boxes
Activated metals	small liners
Other wet wastes (e.g. from West Valley, LWR decommissioning, uranium recycle)	large liners or drums
Other noncompressible equipment, trash, or demolition rubble	large boxes
Other compactible trash	drums
Military wastes	drums

Costs for individual containers are assumed as follows:

Container	Size (ft^3)	Cost ($)
55-gal drum	7.5	25
Small box	16	250
Large box	128	500
Small liner	50	4,000
Large liner	170	5,000

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High Integrity Containers

It has been standard practice in the past to assume no confinement capability following disposal for the containers in which wastes are shipped to disposal facilities. There is little data available, but the data that does exist indicates that there is great variability in the length of time in which the containers retain their form and/or integrity after disposal.

There are many variables that may affect the integrity of currently used waste containers after disposal. These variables include the stability of the waste form (compatibility, resistance to biologic attack, etc.), the void volume of the container (packaging efficiency), the characteristics of the disposal facility site (natural elements such as precipitation and humidity), the depth of disposal (static soil pressures), and the chemical characteristics of the surrounding soils and wastes (corrosiveness). Because of the many unquantifiable and site-specific variables, no attempt has been made in this appendix to estimate and incorporate a confinement capability for typical containers.

However, the concept of a high integrity container (HIC) may be considered as an alternative to waste processing as a means of improving the waste form. In this case, the container would be constructed in a much more robust manner than the containers generally used to transport wastes to disposal facilities. The HIC would be designed to resist crushing from static loads and corrosion from the contained wastes as well as the surrounding soils. The HIC would therefore provide the needed support to disposal cell covers to minimize subsidence and to reduce infiltration. In addition, since the wastes would be contained inside the HIC, leaching of radionuclides from the HIC would be negligible as long as the HIC retained its integrity. (Note that corrosion of a portion of a HIC, which could compromise its ability to withstand leaching, would not necessarily reduce its ability to provide structural support for the disposal cell covers.)

Use of HICs as an alternative to solidification of ion-exchange resins and filter media is allowed by the Part 61 regulation, the South Carolina Department of Health and Environmental Control (the State agency regulating disposal of waste at the Barnwell, S.C. disposal facility), and by the State of Washington. Performance criteria and guidance for HICs have been prepared by NRC and are listed in a Technical Position Paper on waste form (Ref. 12). This guidance is summarized in Table B-6.

Standard HICs

Several HIC designs are currently being marketed. The HICs are constructed principally of polyethylene and are currently available in several sizes. Given adequate lead time for fabricating, special designs are also available upon request. Costs for a HIC are variable and are a function of size, waste form, and other factors. Costs for an empty polyethylene 55-gallon drum HIC appear to be in the range of \$300-400 per HIC. However, costs for larger, liner-sized HICs do not appear to vary significantly depending on size. Polyethylene HICs ranging in size from 80 ft^3 to about 195 ft^3 appear to be in the range of \$5000 per empty container, plus or minus a few hundred dollars. This is about the same range of costs as for carbon steel liners of the same sizes. Additional costs are associated with any internal equipment used for dewatering purposes.

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Table B-6. High Integrity Container Design Guidance

1. Design for 300-year lifetime objective.
2. Design to withstand corrosive and chemical environment.
3. Design to withstand loads associated with disposal.
4. Materials designed to withstand 10^8 Rads.
5. Materials resistant to biodegradation.
6. Design to Department of Transportation Type A package qualification.
7. Conduct prototype testing.
8. Have a quality assurance program.
9. Use a process control program to demonstrate compliance.

These additional costs can range from about $2000 to $4000 per container depending upon the waste media being processed (e.g., resins vs. filter sludge). An additional complication is that quite often HICs are not sold separately but as part of a total service package including processing, shipping, and disposal. Higher costs would be expected for HICs sold separately.

In this report, the following costs for HICs are assumed as a function of container type:

Container	Size (ft^3)	Cost ($)
55-gal drum	7.5	350
Small box	16	4,000*
Large box	128	6,000*
Small liner	50	4,000
Large liner	170	5,000

*Speculation

These costs are uncertain, particularly for small and large boxes. At this time there are no HIC box designs being marketed. If waste stabilization is eventually required, however, for all waste forms, then one option could be emplacement within a HIC. A box shape would have to be developed to be compatible with a number of existing waste compaction equipment designs.

Sealed Source HICs

There is also some interest in developing a high integrity container for disposal of sealed sources and other wastes having high concentrations and very small volumes. One potential design has been suggested by Dornsife for disposal of radium sources (Ref. 13) and is illustrated in Figure B.1. The suggested design is approximately 1.5 inches in diameter and 3.5 inches long, and is constructed of stainless steel pipe. A stainless steel bottom is welded onto the

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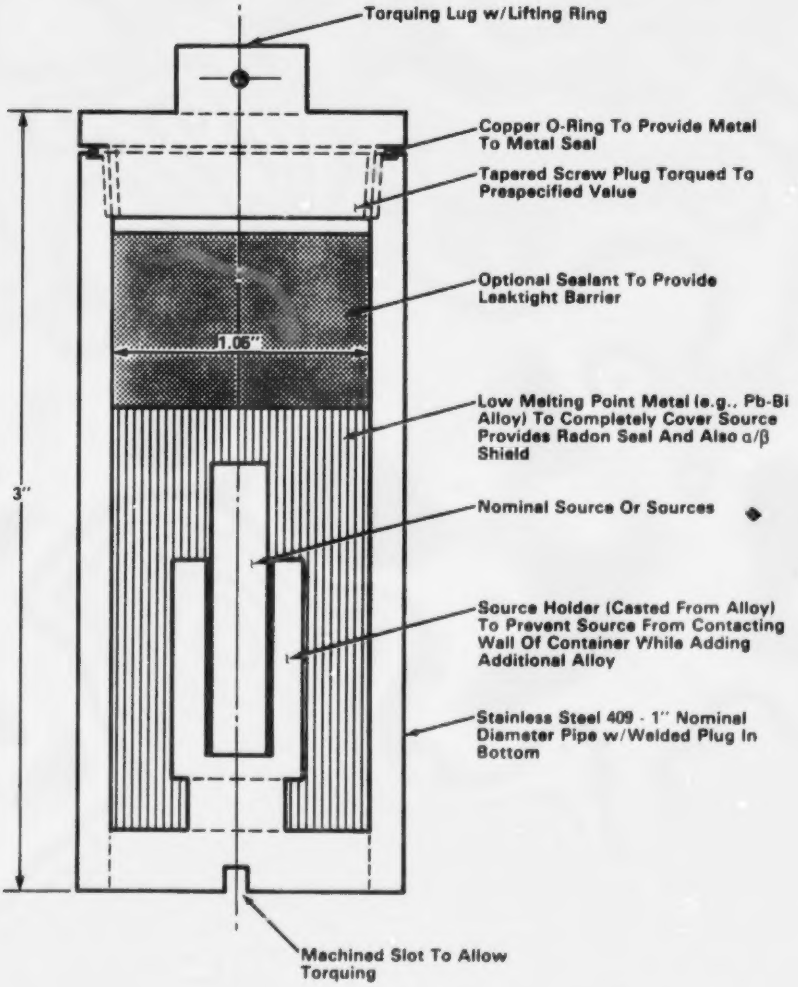
Torquing Lug w/Lifting Ring
Copper O-Ring To Provide Metal To Metal Seal
Tapered Screw Plug Torqued To Prespecified Value
Optional Sealant To Provide Leaktight Barrier
1.06"
Low Melting Point Metal (e.g., Pb-Bi Alloy) To Completely Cover Source Provides Radon Seal And Also α/β Shield
3"
Nominal Source Or Sources
Source Holder (Casted From Alloy) To Prevent Source From Contacting Wall Of Container While Adding Additional Alloy
Stainless Steel 409 - 1" Nominal Diameter Pipe w/Welded Plug In Bottom
Machined Slot To Allow Torquing

Figure B.1. Conceptual Sealed Source High Integrity Container

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bottom of the pipe, and after the source is placed in the container, the remaining void space is filled with a low melting alloy. A tapered screw plug is then torqued on. The container could then be transported to the disposal site within a reusable overpack (e.g., a 6 M/type B shielded container) or solidified within a drum (Ref. 13).

Costs for a sealed source HIC are difficult to estimate. The above container designed by Dornsife is being used in a pilot program to dispose of a very limited number of radium sources. Six stainless steel containers have been constructed by EPA, and are principally being used to ensure safety and stability during handling, transporting, and disposing of the sources. No specific tests have been conducted pursuant to reference 12 to qualify the container as a high integrity container. The costs for construction of the containers was about $30 apiece. Costs would be considerably higher if qualified for use as an HIC (Ref. 14). In this report, a cost of $60/container is assumed.

B.2.4 Waste Process Control Program

The waste form requirements in Section 61.56 of 10 CFR Part 61 require that Class B and Class C waste be disposed in a manner that ensures structural stability. A very common way to meet this requirement is to incorporate the waste into a suitable solidification media. In addition, all licensees must comply with the waste classification requirements in Section 61.55. Additional costs can result from complying with either requirement--in the first case from ensuring that a solidified waste form will meet the stability criteria, and in the second case from determining the concentrations and quantities of specific radionuclides within the waste.

Waste Form Process Control Program

The Branch Technical Position on Waste Form (Ref. 12) provides guidance on acceptable methods for demonstrating stability for Class B and C wastes, and includes guidance on developing and qualifying process control programs for waste solidification processes, designing high integrity containers, packaging filter cartridges, and loading organic ion exchange resins. For waste solidification, different criteria are applicable to different waste classes.

The following technical position is taken (Ref. 12) for solidified Class A waste products:

- Solidified Class A waste products which are segregated from Class B and C wastes should be free-standing monoliths and have no more than 0.5% of the waste volume as free liquids as measured using the method described in ANS 55.1.

- Solidified Class A waste products which are not segregated from Class B and C wastes should meet the stability guidance for Class B and C wastes.

For solidified Class B and C wastes, a series of acceptable methods for demonstrating stability are specified or summarized below (Ref. 12):

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Test	Method	Result
1. Compression strength	ASTM C39 or D621	50 psi
2. Radiation stability		50 psi after 10^8 rads
3. Biodegradation	ASTM G21 and G22	No growth
4. Leachability	ANS 16.1	Leachability index ≥ 6
5. Immersion		50 psi after 90 days
6. Thermal cycling	ASTM B553	50 psi after 30 cycles from -40°C to 60°C
7. Free liquid	ANS 55.1	0.5%
8. Full-scale tests		Homogeneous and correlates lab size sample results

Reference 12 states that an acceptable approach for implementing the tests for solidified Class B and C wastes would be through the qualification of a process control program (PCP). The full battery of tests need only be performed at the time of PCP qualification, although the PCP should be periodically reverified to ensure that the system is operating as designed. The reverification could be accomplished with a short series of tests. The use of generic test data for PCP qualification is acceptable.

Testing may be performed on simulated, non-radioactive samples. If laboratory size specimens are used, these samples should be correlated with full-scale waste forms to demonstrate that the actual waste products will have similar properties as the specimens tested.

Costs. Costs for ensuring waste form stability may be variable depending upon a number of conditions, including the amounts of Class B and C wastes to be solidified, the waste form, and whether installed or vendor-supplied solidification systems are used. A utility, for example, that controls plant processes that only Class A liquids are solidified would not be faced with additional costs for sample testing. A vendor selling a solidification service would spread costs for sample testing and topical report preparation over a number of clients. Conversely, a licensee using installed equipment could have to perform the tests himself and potentially have only a small volume of waste to spread verification costs over. Some of these considerations are discussed in references 15 and 16.

Another consideration is that process formulations that enhance waste stability tend to result in decreased solidification efficiency. That is, assume that a licensee is solidifying liquid waste in order to meet a free-standing monolith requirement. However, if the waste form must be structurally stable, less liquid waste (and more solidification agent) is used, since the optimum waste/binder ratio for structural stability is generally less than the maximum waste/binder ratio that can be achieved.

As a rule of thumb, it is assumed that unit costs for waste solidification (\$ per m^3 of input waste) are increased by 20% for improved process control programs which achieve the Section 61.56 stabilization criteria.

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Waste Classification Process Control Program

The waste classification technical position (Ref. 23) describes overall procedures acceptable to the NRC staff which may be used by licensees to determine the classification concentrations of radionuclides listed in Section 61.55. Several of these radionuclides are difficult to assay, although they are judged to be important from a disposal viewpoint and there is therefore a need to know the inventories of these radionuclides upon closure of a disposal facility. The technical position is intended to provide guidance on practical methods for performing assays and determining radiouclide quantities (Ref. 23):

The waste classification technical position discusses four acceptable methods for classifying wastes (Ref. 23):

- materials accountability,
- classification by source,
- gross radiation measurements, and
- direct measurement.

The materials accountability method would be generally used by institutional and industrial waste generators who receive known quantities of specific nuclides. By knowing how much activity is used in an experiment, the radionuclide quantities can be computed. Classification by source is an extension of the materials accountability method. An example of this method might be a university who has varied experiments underway using several nuclides in separated laboratories.

Gross radiation measurements can be applied to heterogeneous materials, such as trash, if the gross survey reading can be correlated to a measured nuclide distribution. An example of this method might be a power plant which has obtained a nuclide distribution for trash by assaying a consolidated series of smears taken from inside the facility.

The method of direct measurement allows the use of scaling factors for determining quantities of the difficult-to-measure nuclides in the wastes. In applying this method a waste generator would analyze samples of the major waste streams at the facility. From this analysis the generator would develop scaling factors between easily measured nuclides using gamma spectroscopy and those which are difficult to directly measure. An example of such a scaling method is the Ce^{144} -$Pu^{239,240}$ correlation which is currently used at several power facilities. In this case, chemically similar nuclides have been correlated and scaling factors computed. Using this method reference 23 recommends updating the correlation factors by waste stream reanalysis on an annual basis for Class B and C wastes and on a biennial basis (every 2 years) for Class A wastes.

Costs. Costs for implementation of the waste classification requirements principally entail (1) some additional paperwork, and (2) determining radionuclide concentrations and quantities within waste streams. Both of these tasks are believed to be significantly more involved for fuel cycle licensees--particularly nuclear power plants--than for non fuel cycle licensees. The latter licensees generally handle a known quantity of radionuclides, and are able to determine inventories within particular waste containers as part of normal bookkeeping procedures.

Estimates of waste classification costs for nuclear power plants are provided in reference 15. These estimated costs total approximately $86,000 per plant for the first year (for sampling, establishing scaling factors, etc.), plus about $38,000 per plant for subsequent years. This is an average of about $40,000 per plant per year.

B.3 WASTE SPECTRA

This section describes the five waste spectra that are used in this report. The concept "waste spectrum" as used here denotes the total volume and properties of the waste streams after they have been processed by a set of selected waste treatment options. Each spectrum corresponds to a general level of waste performance in terms of waste stability, resistance to wind mobilization, resistance to leaching, and physical, chemical, and radiological properties that can be achieved by establishing operational and/or administrative requirements. The spectra differ significantly in waste volumes, radioactive concentrations, and performance.

A total of four waste spectra were devised for the original Part 61 impact analysis methodology (Refs. 1, 2), and subsequently used in the draft Part 61 EIS (Ref. 3). Two additional waste spectra were added for the final Part 61 EIS (Ref. 17), making a total of six.

The focus of this updated impact analysis methodology is somewhat different than that for the original methodology. There is less interest than illustrating the problems associated with past or poor waste processing and packaging practices, and more interest in analyzing waste processing, packaging, and disposal practices against an existing regulatory standard (10 CFR 61). There is also somewhat less interest in volume reduction as an object of analyses. As a result, the number of waste spectra has been reduced from six to five. This has been accomplished by combining former waste spectra 6 and 1 as used in the final Part 61 EIS (Ref. 17).

B.3.1 Waste Spectra Descriptions

The radioactive concentrations of each waste stream for each spectrum depends on the change in the volume of the stream during processing. Whenever a process is applied to a waste stream that results in volume reduction of the stream, its concentrations are increased accordingly. Similarly, whenever a process is applied that results in a volume increase, the concentrations are decreased accordingly. The minute quantities of radionuclides that are lost during these processes (e.g., the radionuclides may become attached to the process vessel walls) have been conservatively neglected.

The five waste spectra are used to consider the range in waste performance that can be achieved through alternative operational and/or administrative requirements. With each of the first four respective spectra, increased waste processing is assumed. This results in waste forms having greater stability, better leaching characteristics, lessened dispersibility, lower volumes, and higher concentrations. The effect of these alterations in waste form and radionuclide concentrations on radiological impact measures such as groundwater migration and to exposures to potential inadvertent intruders may be compared against costs and other nonradiological impact measures.

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In developing the spectra, it was recognized that a considerable amount of change is currently taking place in existing waste processing and packaging techniques. This relatively rapid change makes it difficult to characterize current waste processing and packaging practices. For example, due to the rise in costs for waste disposal, there is increased use of volume reduction procedures (such as use of compactors) by waste generators. In addition, license conditions at two disposal facilities are requiring that resins and filter media be either solidified or placed into high integrity containers prior to disposal.

Therefore, the first two spectra were established to more or less straddle existing practice. Waste spectrum 1 represents existing or past practices, and characterizes a condition in which compressible waste streams from large waste generators are subject to compaction, while high activity waste streams are disposed in an unstable form. Waste spectrum 2 characterizes improvements in the form of the waste through processing with relatively modest expenditures of time and money. Waste spectra 3 and 4 represent more extreme waste processing and packaging practices, including (waste spectrum 4) extensive incineration of combustible waste streams. Waste spectrum 5 characterizes (for most waste streams) use of containers providing structural support to achieve waste form stability rather than processing to a solid form.

The general assumptions made for these waste spectra are presented below.

B.3.1.1 Waste Spectrum 1

This waste spectrum represents a continuation of past or existing waste management practices. It characterizes waste characteristics projected to result without requirements for waste stability and considering the increasing costs for waste disposal. Some waste streams such as LWR concentrated liquids are solidified, although the process control programs for the waste forms are only oriented toward meeting a free-standing solid requirement rather than a structural stability requirement. The following general assumptions are made.

Nuclear Power Plants

LWR resins (P-IXRESIN, B-IXRESIN) and filter sludges (P-FSLUDGE, B-FSLUDGE) are shipped to disposal facilities in a dewatered form. LWR concentrated liquids (P-CONCLIQ, B-CONCLIQ) are concentrated in accordance with current practices, and are solidified in cement (solidification scenario A). PWR cartridge filters (P-FCARTRG) are packaged for shipment by placing the filters within a 55-gallon drum. The resulting void spaces within the drum result in a structurally unstable waste form. Combustible trash waste streams (P-COTRASH, B-COTRASH) are compacted using commonly available equipment, resulting in an average volume reduction across all plants of about 3 for PWRs and 2 for BWRs. Noncombustible trash waste streams (P-NCTRASH, B-NCTRASH) are assumed to contain too much contaminated metals and other materials which are difficult to compact, and so these waste streams are packed into containers without any additional processing.

LWR activated metal core components (L-NFRCOMP) are assumed to be packaged into containers with the interstitial spaces in the packages either left as voids or filled with material such as compressible waste forms. This packaging practice results in an unstable waste form. Liquid decontamination solutions from LWR primary coolant decontamination operations (L-DECONRS) are solidified in a synthetic polymer (solidification scenario B). As for LWR concentrated liquids, the solidification process control program is assumed to be sufficient to achieve a free-standing solid, but not sufficient to comply with the waste form structural stability requirements in 10 CFR Section 61.56.

Other Nuclear Fuel Cycle Facilities

Compressible trash from uranium fuel fabrication facilities (F-COTRASH) are compacted using standard techniques, while fabrication and conversion process wastes are packaged for shipment with little further processing (F-PROCESS, U-PROCESS), as is noncompressible trash (F-NCTRASH). Waste from mixed oxide facility decontamination (L-PUDECON) and from spent fuel chemical analyses (L-BURNUPS) is characterized as a mixture of solidified liquids (especially L-BURNUPS), metal, glass, paper, etc., and is treated (for packaging purposes) like noncompressible trash.

Institutional Waste

These wastes include trash, liquids, scintillation media, and biological material generated by both large and small institutional facilities. Trash from large facilities (I-COTRASH) is assumed to be compacted prior to shipment, while for small facilities, compaction equipment is judged to be generally too expensive to justify the small quantity of trash waste generated per facility. Hence, little further processing is assumed for the I+COTRASH stream. For both small and large facilities, liquids (ABSLIQD), liquid scintillation media (LIQSCVL) and biological material (BIOWAST) are packed into containers using absorbent material to reduce the potential for free liquids. This increases the volume of the final waste product.

Industrial Waste

Very little additional waste processing is assumed for industrial waste streams. Some compaction is performed for compressible trash waste streams generated by large generators (N-SSTRASH, N-LOTRASH). Waste from large radioisotope manufacturers is difficult to characterize. It typically contains a mixture of materials such as solidified liquids, microspheres, failed equipment, organic and flammable material, and so forth. The N-ISOPROD waste stream also contains some activated metals, although most of the activity is contained within cement solidified liquids. As in the L-BURNUPS stream, liquids are solidified in small (e.g., 1-gallon containers) which are placed with other material into larger containers such as 55-gallon drums.

Waste from large tritium and carbon-14 generators is a mixture of materials, including trash, glass, tritium gas, and absorbed and solidified liquids. The absorbed liquids are composed of organic rather than aqueous liquids. Compaction is applied to the N-NECOTRA waste stream, while the N-NEABLIQ, N-NEVIALS, and N-MWABLIQ waste streams are absorbed onto absorbents within drums. The N-NESOLIQ and N-MWSOLIQ waste streams are solidified within cement. The N-NETRGAS, N-NETRILI, and N-NECARLI waste streams are packaged within special high integrity containers.

High activity liquid scintillation waste (N-TRISCNT) and aqueous liquids (N-TRILIQD) are packed with absorbent material into drums. Little or no processing is performed for other waste streams from small manufacturers using tritium.

The N-HIGHACT waste stream is packaged in a similar manner as activated metals from nuclear power plants. The waste is packaged into containers with the interstitial spaces in the container either left as voids or filled with material such as compressible trash. This packaging practice results in an unstable waste form. Source waste streams are assumed to be disposed by placement within drums using other material such as contaminated trash as shielding. Higher activity gamma sources are first placed within lead shipment pigs prior to packing within drums.

Other Non Fuel Cycle Waste

These waste streams include radium sources, radium contaminated resins (N-RARESIN), and military waste. Sources are disposed by placement within containers containing other material such as trash. Resins are shipped to disposal in a dewatered form. Military waste is a composite of materials. The M-NAVYWET waste stream is mostly liquids solidified in cement, while the M-NAVYDRY waste stream is treated like noncompactible trash.

Non-Routine Waste

These waste streams include those from uranium fuel reprocessing, mixed oxide fuel fabrication, nuclear power plant decommissioning, and the West Valley Demonstration Project. Wastes from these activities are processed and packaged for shipment in a parallel manner as similar wastes from nuclear power plants.

Liquid high-level waste from uranium fuel reprocessing is vitrified in glass (R-HLLWFRP). Very little processing is performed for fuel hardware and hulls. Liquids (R-ILLWFRP, R-SBCOLIQ) are solidified in cement, while resins and sludges (R-MPRESIN, R-SBRESIN) are dewatered. Compressible trash (R-MPCOTRH, R-MPCOTRL, R-SBCOTRA, R-UFCOTRA, R-PUCOTRA) is compacted using equipment currently in common use. Noncompactible trash (R-MPNCTRA, R-SBNCTRA), R-UFNCTRA, R-PUNCTRA) is disposed with little further processing, as are fines and mud from the UF$_6$ conversion process (R-UFK2MUD, R-UFFINES). Degraded extractant is assumed to be absorbed.

Compressible waste from mixed oxide fuel fabrication (R-MOXCOTR) is compacted using commonly available equipment while liquids (R-MOXSOLN) are solidified in cement (solidification scenario A).

Wastes from nuclear power reactor decommissioning parallel those from nuclear power plant operations. Activated metals may of themselves meet the stability requirements in 10 CFR Section 61.56 (DECORES, DEACINT, DEACVES), although the method of packaging leads to voids in the disposed waste. Activated and contaminated concrete rubble are disposed without further processing, as are contaminated metals (DECONME). Compressible trash (DETRASH) is compacted. As for wet wastes, resins are disposed in a dewatered form (DERESIN), evaporator bottoms (DEEVAPB) are solidified in cement (solidification scenario A), and PWR filter cartridges (P-DEFILCR) are placed into a drum in an unstable manner.

A number of waste streams result from the West Valley Demonstration Project. Of these, the two high-level liquid waste streams are vitrified in glass (W-THORHLW, W-PUREHLW) while other miscellaneous liquids are solidified in cement (solidification scenario A). These include the radwaste treatment system (RTS) and presolidification decontamination (PD) liquids, decontamination and decommissioning (DD) liquids (W-DDLTLQD, W-DDHTLQD), and various liquid wastes from vitrification. Resins and sludges from various activities (W-LLWTFRE, W-FRSRESN, W-RTSRESN, W-DDRESIN, W-VITZEOL) are shipped in a dewatered form. Little is done to process noncompactible wastes (W-NCSOLID), failed contaminated equipment (W-HTEQUIP, W-LTEQUIP), old fuel storage racks (W-DDRACKS), and demolition rubble (W-DDLTRUB, W-DDHTRUB).

B.3.1.2 Waste Spectrum 2

This waste spectrum is very similar to waste spectrum 1. Three changes are especially significant, however. First, liquids which were originally solidified in waste spectrum 1 are also assumed to be solidified in this waste spectrum, and also using the same solidification media as before. (Except for the L-DECONRS waste stream, all were solidified using solidification scenario A.) However, solidification is generally performed using a process control program that is in accordance with the Low-Level Waste Licensing Branch Technical Position on Waste Form (Ref. 12) or by some other method that complies with 10 CFR Section 61.56. Second, ion exchange resins and filter sludges are solidified rather than being disposed in a dewatered form. Third, activated metal wastes are disposed in a stable manner. This is assumed to be accomplished by filling interstitial spaces within waste containers with an inert low compressible material such as sand.

Nuclear Power Plants

This spectrum is the same as waste spectrum 1 except that ion exchange resins and filter media are solidified (solidification scenario A) in addition to other liquids. The L-DECONRS stream is again solidified using solidification scenario B. PWR filter cartridges are solidified within 55-gallon drums. This is accomplished by filling the voids in the drums without appreciably increasing the shipped volume of the waste stream. The activated metal waste stream (L-NFRCOMP) is stabilized for disposal using improved packaging techniques as discussed above.

Other Nuclear Fuel Cycle Facilities

No change from waste spectrum 1 is assumed. (Also see Industrial Waste, below, in relation to the L-BURNUPS waste stream.)

Institutional Waste

No change from waste spectrum 1 is assumed.

Industrial Waste

No change from waste spectrum 1 is assumed for source and special nuclear material trash and other miscellaneous waste, or for other low activity trash and waste. For N-ISOPROD waste, of which a significant portion is solidified

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liquids, use of an improved solidification process control program is not implemented. (A similar situation was assumed above for the L-BURNUPS waste stream.) This is because the technique of packaging the solidified waste in the same container with miscellaneous other materials still results in an unstable waste form, making any improvements in solidification pointless from a stability point of view. Therefore, no change from waste spectrum 1 is assumed for wastes from large radioisotope manufacturers. No change is assumed for large tritium and carbon-14 manufacturers, with the exception that an improved process control program is assumed for solidification of the N-NESOLIQ and N-MWSOLIQ waste streams. No change is assumed for small manufacturers using tritium, with the exception of the liquid waste stream (N-TRILIQD) which is solidified using an improved process control program.

The N-HIGHACT waste stream is assumed to be stabilized as discussed above for the L-NFRCOMP waste stream. Sealed sources are assumed to be stabilized by placing the source in the middle of a 55-gallon drum filled with cement. The solidification process is assumed to be conducted in a manner so that the waste meets the structural stability requirements of 10 CFR Section 61.56. This should be fairly simple given the ease in which the chemical constituents within the cement can be controlled (waste liquids are not being solidified).

Other Non-Fuel Cycle Waste

Radium sources are assumed to be solidified in cement as discussed above for other waste sources. Radium contaminated resins are solidified (solidification scenario A), while an improved process control program is assumed for the M-NAVYWET stream.

Non-Routine Waste

For uranium fuel reprocessing and mixed oxide fuel fabrication, the only changes include solidification of resins and filter sludges, use of improved solidification process control programs, and improved packaging for activated metals. A similar situation exists for wastes from nuclear power plant decommissioning and for the West Valley Demonstration Project. Cartridge filters from decommissioning pressurized water reactors (P-DEFILCR) are stabilized in the same manner as cartridge filters from normal operations (P-FCARTRG).

B.3.1.3 Waste Spectrum 3

This waste spectrum is very similar to waste spectrum 2 except that an improved solidification media is assumed to be used (solidification scenario B) and increased compaction is performed on compressible waste. Liquids generated by large facilities are also generally subjected to increased volume reduction. The solidification process control program is generally such that the waste form meets the Part 61 stability requirements. Compaction at large facilities is assumed to be generally accomplished using improved compactor/shredders which achieve greater volume reduction than most of the equipment being used today. Compressible waste from a number of small facilities--particularly small institutional and industrial facilities--are assumed to be generally compacted at a centralized operation assumed to be operated as an adjunct to the disposal facility.

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Nuclear Power Plants

LWR process wastes, including PWR filter cartridges, are solidified assuming use of solidification scenario B as part of an improved process control program. LWR concentrated liquids are first evaporated to 50 weight percent solids using evaporator/crystallizers. Compactible trash streams are compacted using improved compactor/shredders.

Other Nuclear Fuel Cycle Facilities

The F-COTRASH waste stream is subjected to improved compaction. The only other change is for the L-BURNUPS waste stream, for which solidification scenario B is used rather than solidification scenario A. As for waste spectrum 2, the improved solidification media does not result in a stable waste form.

Institutional Waste

In this spectrum, trash waste streams are subjected to improved compaction either by the generator (I-COTRASH) or at the disposal facility (I+COTRASH). Liquid scintillation vials are assumed to be crushed using currently available equipment and packaged in absorbent material. This can occur either at the waste generating facility or at the disposal site.

Industrial Waste

Improved compaction as outlined above is implemented for the N-SSTRASH, N+SSTRASH, N-LOWASTE, and N+LOWASTE streams. For large radioisotope manufacturers, the only change is that solidification scenario B is assumed for the N-ISOPROD waste stream. This once again does not result in a stable waste form. Solidification scenario B is assumed for the N-NESOLIQ and N-MWSOLIQ waste streams. For small manufacturers using tritium, scintillation vials are compacted at the disposal facility using standard compaction techniques. Tritium liquids are solidified using solidification scenario B. Sealed sources are solidified in drums using cement.

Other Non Fuel Cycle Waste

Radium sources are solidified in drums using cement. Radium-contaminated resins and the M-NAVYWET stream are also solidified using solidification scenario B.

Non-Routine Waste

These waste streams are handled like those from nuclear power reactors. Liquids and evaporator bottoms are evaporated to 50 weight percent solids using improved evaporator/crystallizers, all wet wastes are solidified using solidification scenario B, and compressible trash waste streams are compacted using improved compaction techniques.

B.3.1.4 Waste Spectrum 4

This waste spectrum is devised assuming extreme volume reduction--i.e., about the maximum volume reduction that can currently be theoretically achieved. All wastes amenable to evaporation or incineration with fluidized bed technology (e.g., LWR process wastes) are calcined and then solidified using solidification scenario C. Institutional and industrial waste streams at large facilities are incinerated using a pathological incinerator, while several institutional and industrial waste streams are shipped to a large fluidized bed incinerator assumed to be located at the disposal facility. Noncompactible and contaminated metal waste streams are also shipped to the disposal facility where they are compacted using a large hydraulic press. Sources are solidified in drums using cement.

B.3.1.5 Waste Spectrum 5

This spectrum incorporates for most waste streams high integrity containers (HICs) to achieve a stable waste form. Relative to waste spectrum 1, most waste streams (other than activated metals) which had previously been in an unstable form are stabilized using HICs. Activated metals are stabilized by filling interstitial voids in a waste container with a noncompressible material. LWR concentrated liquids are solidified assuming solidification scenario A procedures, while waste from medical isotope production facilities is assumed to be solidified using solidification scenario B. A process control program compatible with the waste form BTP is implemented in all cases. Wastes from tritium manufacturing facilities are also placed into HICs, as are sealed sources.

B.3.2 Spectra Data File Components

For each of the five waste spectra, a data file was constructed consisting of five major groups of waste form and packaging parameters:

- Waste form behavior indices (seven indices total);
- Waste processing parameters;
- Volume reduction and volume increase factors; and
- Waste densities.

These groups of parameters are discussed in this section.

B.3.2.1 Waste Form Behavior Indices

The effects of different waste physical and chemical characteristics must be included in the impact analyses. One such tool is quantifying these characteristics through discrete indices that trigger specific computational procedures in the impacts analyses. This is the approach adopted in this report.

The characteristics important in determining the effects of different waste behavior include the flammability of the waste form at the time of disposal, the dispersibility of the waste form several decades after disposal, the resistance of the waste form to leaching, the accessibility of the radionuclides to transfer agents such as wind or water, the relative mobility of the radionuclides (the presence or absence of chelating agents or organic chemicals), the structural stability of the waste, whether the waste form is composed of activated metal, and whether the waste form is a sealed source or some other small and highly concentrated material. These seven properties are quantified through seven waste form behavior indices defined in Table B-7 and discussed below.

Table B-7. Waste Form Behavior Indices

Parameter	Symbol	Indices
Accident/Scatter	(I4/ISC)	0 = severe 1 = moderate 2 = slight to moderate 3 = low
Accident/Flammability	(I4/IFL)	0 = flammable (supports burning) 1 = burns if heat supplied (does not support burning) 2 = low flammability (mixture of material with indices of 0 and 2) 3 = nonflammable
Dispersibility	(I5)	0 = severe 1 = moderate 2 = slight to moderate 3 = low
Leachability	(I6)	1 = unsolidified waste form 2 = solidification scenario A 3 = solidification scenario B 4 = solidification scenario C
Chemical Content	(I7)	0 = no chelating agents or organic chemicals 1 = chelating agents or organic chemicals are likely to be present in the waste form
Stability	(I8)	0 = structurally unstable waste form 1 = solidified waste form 2 = structurally stable solidified waste form 3 = stabilized by placement into HIC 4 = stabilized by another means
Activated Metal	(I9)	0 = not activated metal waste 0 < activated metal waste
Sources	(I10)	0 = not source waste 0 < source waste

Many of the waste form behavior indices rank particular waste forms in terms of tendencies to behave in a certain way. For example, the flammability component of the accident index compares different waste forms in terms of their tendency to burn, the dispersibility index compares different waste forms in terms of their tendency to disperse into the air, and so forth. Many of the integer values assigned to these rankings are done so in a rather subjective way.

However, this is not believed to be a major problem since the major purpose of the impact analysis methodology is to enable the user to compare, and quantify to a certain extent, trends associated with particular alternative rulemaking actions. The waste form behavior indices should not be used as a substitute for real data and test results for specific waste streams.

The accident index (I4) is comprised of two subindices which relate to the tendency for waste forms to disperse into the air during hypothetical operational accident scenarios. The two subindices include the scatter index (ISC) and the flammability index (IFL).

The scatter index (ISC) qualitatively ranks waste forms based on their relative tendency to disperse into the air during a potential operational accident in which a waste container is violently damaged. A hypothetical situation could include a waste container potentially dropped from a significant height. This subindex is similar to the following dispersibility index (I5).

Waste forms which are judged to have a low probability of becoming dispersed during a severe accident are assigned an index of 3. Those waste forms which have a high potential for becoming dispersed during a severe accident are assigned an index of 0. Waste forms that are judged to potentially crumble or fracture extensively during a severe accident are assigned an index of 1. Waste forms estimated to consist of a mixture of materials with scatter indices of 0 and 1 are assigned an index of 2.

This index is difficult to assign rigorously given the lack of comparative data. However, the following general index assignments were typically made.

ISC	Dispersion Tendency	Waste Forms
3	low	Waste solidified in synthetic polymers; activated metals; solidified filter cartridges; sealed sources.
2	slight to moderate	Waste solidified in cement; filter cartridges; activated concrete.
1	moderate	Compactible trash; noncompactible trash; failed equipment; contaminated metal; contaminated concrete; biological waste; building rubble; dewatered ion exchange resins.
0	severe	Dewatered sludge; ash; dirt; powders; absorbed liquids; scintillation media; uranium conversion and fabrication process waste; gas.

The flammability index (IFL) ranks waste forms according to their flammability prior to disposal. Waste forms that will not burn even on prolonged exposure to open flame and moderately intense heat are assigned an index of (3). Those waste forms that will sustain combustion are assigned an index of (0). Between these extremes are two additional flammability categories. Waste forms that will ignite but will not sustain burning under these conditions are assigned an index of (1). Waste forms consisting of a mixture of materials with flammability indices (3) and (1) are assigned an index of (2).

Based upon reference 1 and other considerations, the following general index value assignments have been made:

IFL	Flammability Tendency	Waste Forms
3	nonflammable	Activated metals; liquids and other wastes solidified in cement; sealed sources; uranium conversion and fuel fabrication process wastes.
2	low flammability	Dewatered sludge; calcined material solidified in synthetic polymers.
1	burns if heat supplied	Dewatered ion exchange resins; unsolidified filter cartridges; materials solidified in synthetic polymers; noncompactible trash; absorbed liquids.
0	flammable	Combustible trash; liquid scintillation media.

The dispersibility index (I5) is a qualitative measure of the potential for suspension of radioactivity, should the waste form be exposed to wind after a significant period (on the order of 100 years). Waste forms which are judged to have a low probability of becoming suspended into respirable particles are assigned an index of (3). Those waste forms that have a high potential of becoming suspended into respirable particles are assigned an index of (0). Waste forms that tend to crumble or fracture extensively and those subject to relatively rapid (within about 100 years) decomposition are assigned an index of (1). Waste forms consisting of a mixture of materials with dispersibility indices of (3) and (1) are assigned an index of (2).

General assumptions regarding the index values assigned to various waste forms include the following:

I5	Dispersibility Tendency	Waste Forms
3	low	Activated metals; waste solidified in synthetic polymers with improved process control program.
2	slight to moderate	Waste solidified in cement with improved process control program; waste solidified in synthetic polymers with minimum process control program.
1	moderate	Waste solidified in cement with minimum process control program; unsolidified filter cartridges; compactible trash; noncompactible trash; dewatered ion exchange resins.
0	severe	Dewatered filter sludge; absorbed liquids; liquid scintillation media; biological waste; uranium fabricaion and conversion process wastes.

The leachability index (I6) is a qualitative measure of the waste form's resistance to leaching and is determined by the solidification procedures used. Unsolidified waste forms, which are assumed to be readily leached, are assigned an index of (1). Solidification scenarios A, B, and C (discussed in Section B.2.2) are assigned indices of 2, 3, and 4, respectively.

The chemical content index (I7) denotes whether the waste form may contain significant quantities of chelating agents or organic chemicals that may increase the mobility during leaching and subsequent migration from the disposal cell. An index value of (0) indicates a likelihood that significant quantities of these chemicals or agents are absent, whereas an index value of (1) indicates a likelihood of their presence.

The stability index (I8) denotes whether the waste form is likely to reduce in volume after disposal due to compressibility, large internal void volumes, and/or chemical and biological attack. With the exception of waste streams packaged in high integrity containers, no credit is taken for the waste containers. An index value of (0) indicates the likelihood of structural instability, whereas a value greater than 1 indicates a structurally stable waste form.*

An index value of (1) indicates a waste form that has been solidified without use of a process control program that ensures compliance with the waste form structural stability requirements in 10 CFR 61.56. In this case, waste is processed to the extent that it achieves a free-standing form but does not necessarily meet the stability requirements. An acceptable program for ensuring compliance with the stability requirements is outlined in reference 12 and waste streams assumed to be solidified in accordance with reference 12 are assigned an index value of (2).

An index value of 3 indicates that stability is achieved through placement of the waste into a high integrity container (HIC). An index value of 4 indicates a waste stream stabilized by other means. An example would be a large metal component which would normally be considered "inherently" stable but has been placed into a package so that voids exist between the component and the package. The package containing activated metals could be stabilized by filling the interstitial voids with an inert incompressible material such as sand.

The activated metal index (I10) denotes a correction factor in the analyses for activated metals. Surface contaminated wastes and waste containing radioactivity in readily soluble forms are assigned an index of (0). Most waste streams fall into this category. The waste forms that are almost exclusively activated metals with imbedded radioactivity not readily accessible to the elements are assigned an index of (1) or higher.

The specific value of the index is used to access information (from a file called METALS.DAT) on the interaction of the activated metal waste stream with the environment. This information includes factors related to: (1) the time period (in years) it will take for the waste stream to corrode completely, (2) the air dispersibility of the corrosion products, (3) the water solubility of the corrosion products, and (4) the self shielding provided by the waste stream against direct radiation. (Given the current lack of data, air dispersibility and water solubility factors are conservatively taken to be unity.)

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*The assumptions regarding waste stability are made here in order to enable overall assessments regarding economic and radiological impacts of implementing various alternative waste form requirements. Assessments regarding specific individual wastes and waste forms, however, should be made on a case basis in compliance with 10 CFR Part 61 and reference 12.

The source index (I10) denotes whether (I10 > 0) or not (I10 = 0) the waste stream has a form such as a sealed source in which radionuclides are contained in a very small volume. Source waste streams are considered in a different manner from other waste streams in that the total activity per source, rather than the specific concentration, is used for waste classification purposes.

All waste streams for which radionuclide content is expressed as a concentration (activity per unit volume or mass) are assigned an I10 index of (0). Most waste streams fall into this category. Any other positive index is used to indicate a waste stream for which the radionuclide content is specified in terms of total activity per source. The specific value of this index is used to access information (from a file called SOURCE.DAT) on the interaction of the source waste stream with the environment. This information includes factors related to: (1) the time after closure (in years) during which the source can be assumed to be inaccessible to transfer agents, (2) the air dispersibility of the source, (3) the water solubility of the source, and (4) the self shielding provided by the waste stream against direct radiation. (Given the current lack of data, the air dispersibility factor and water solubility factors are both conservatively assumed to be unity.)

Waste behavior indices that have been assumed for each of the waste streams for each of the five waste spectra are presented in Table B-8.

B.3.2.2 Processing Impacts

Processing impacts in addition to those associated with treatment operations performed in waste spectrum 1 include population and occupational exposures, and costs.

Population impacts from processing depend primarily on the radioactive contents of the waste streams and secondarily on the location at which the processing takes place. Only incineration (pathological incinerators and incinerator/calciners) is assumed to result in a release of radioactivity which could result in potentially significant additional population exposures. Occupational exposures depend on the environment in which the waste processing is being performed in addition to the waste activity. The cost of waste processing also depends on the size of the facility as well as the specific process being utilized. In addition, packaging characteristics of a waste stream are specified in order to calculate exposures associated with waste loading, transportation, and emplacement at the disposal facility.

In order to account for these variations, six indices have been assigned to each waste stream in each spectrum and are utilized in the calculation of waste processing impacts. The values of these indices trigger specific calculational procedures in the calculation of the impact measures. The six indices are combined into a single large index called the waste processing parameters index (I1). Meanings assigned to each of the indices are summarized in Table B-9 and are further discussed below.

Table B-8. Waste Form Behavior Indices for Waste Spectra 1-5

Waste Stream		Waste Spectrum 1							Waste Spectrum 2							Waste Spectrum 3							Waste Spectrum 4							Waste Spectrum 5						
Name	No.	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10
P-IXRESIN	1	11	1	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	11	1	1	0	3	0	0
P-CONCLIQ	2	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
P-FSLUDGE	3	02	0	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	02	0	1	0	3	0	0
P-FCARTRG	4	21	1	1	0	0	0	0	33	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	31	1	1	0	3	0	0
B-IXRESIN	5	11	1	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	11	1	1	0	3	0	0
B-CONCLIQ	6	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
B-FSLUDGE	7	02	0	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	02	0	1	0	3	0	0
P-COTRASH	8	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
P-NCTRASH	9	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	3	4	0	0	0	0	11	1	1	0	3	0	0
B-COTRASH	10	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
B-NCTRASH	11	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	3	4	0	0	0	0	11	1	1	0	3	0	0
L-NFRCOMP	12	33	3	1	0	0	1	0	33	3	1	0	4	1	0	33	3	1	0	4	1	0	33	3	1	0	4	1	0	33	3	1	0	4	1	0
L-DECONRS	13	11	1	1	1	0	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	32	3	4	0	2	0	0	11	1	1	1	3	0	0
F-PROCESS	14	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	3	0	0
F-COTRASH	15	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
F-NCTRASH	16	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
U-PROCESS	17	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	3	0	0
L-PUDECON	18	11	1	1	1	0	0	0	11	1	1	1	0	0	0	11	1	1	1	0	0	0	11	1	1	1	0	0	0	11	1	1	1	3	0	0
L-BURNUPS	19	12	1	2	0	1	0	0	12	1	2	0	1	0	0	11	1	3	0	1	0	0	21	1	4	0	1	0	0	12	1	2	0	3	0	0
I-COTRASH	20	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
I+COTRASH	21	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
I-ABSLIQD	22	01	0	1	1	0	0	0	01	0	1	1	0	0	0	01	0	1	1	0	0	0	32	3	4	0	2	0	0	01	0	1	1	3	0	0
I+ABSLIQD	23	01	0	1	1	0	0	0	01	0	1	1	0	0	0	01	0	1	1	0	0	0	01	0	1	1	0	0	0	01	0	1	1	3	0	0
I-LIQSCVL	24	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	32	3	4	0	2	0	0	00	0	1	1	3	0	0
I+LIQSCVL	25	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	3	0	0
I-BIOWAST	26	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	32	3	4	0	2	0	0	11	0	1	1	3	0	0
I+BIOWAST	27	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	3	0	0
N-SSTRASH	28	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	32	3	4	0	2	0	0	11	1	1	0	3	0	0
N+SSTRASH	29	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	32	3	4	0	2	0	0	11	1	1	0	3	0	0
N-SSWASTE	30	13	0	1	0	0	0	0	13	0	1	0	0	0	0	13	0	1	0	0	0	0	13	0	1	0	0	0	0	13	0	1	0	3	0	0
N-LOTRASH	31	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
N+LOTRASH	32	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
N-LOWASTE	33	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	3	0	0
N-ISOPROD	34	11	1	2	0	1	0	0	11	1	2	0	1	0	0	21	2	3	0	1	0	0	31	2	4	0	1	0	0	11	1	2	0	3	0	0
N-ISOTRSH	35	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	3	0	0
N-SORMFG1	36	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
N-SORMFG2	37	11	1	1	1	0	0	0	11	1	1	1	0	0	0	11	1	1	1	0	0	0	11	1	1	1	0	0	0	11	1	1	1	3	0	0
N-SORMFG3	38	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
N-SORMFG4	39	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
N-NECOTRA	40	10	1	1	1	0	0	0	10	1	1	1	0	0	0	10	1	1	1	0	0	0	32	3	4	0	2	0	0	10	1	1	1	3	0	0
N-NEABLIQ	41	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	10	0	1	1	3	0	0
N-NESOLIQ	42	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
N-NEVIALS	43	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	3	0	0
N-NENCGLS	44	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	3	0	0
N-NEWOTAL	45	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	0	0	0	11	0	1	1	3	0	0
N-NETRGAS	46	13	0	1	0	3	0	0	13	0	1	0	3	0	0	13	0	1	0	3	0	0	13	0	1	0	3	0	0	13	0	1	0	3	0	0
N-NETRILI	47	20	0	1	1	3	0	0	20	0	1	1	3	0	0	20	0	1	1	3	0	0	20	0	1	1	3	0	0	20	0	1	1	3	0	0
N-NECARLI	48	20	0	1	1	3	0	0	20	0	1	1	3	0	0	20	0	1	1	3	0	0	20	0	1	1	3	0	0	20	0	1	1	3	0	0
N-MWTRASH	49	10	1	1	1	0	0	0	10	1	1	1	0	0	0	10	1	1	1	0	0	0	10	1	1	1	0	0	0	10	1	1	1	3	0	0

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Table B-8. (continued)

Waste Stream		Waste Spectrum 1							Waste Spectrum 2							Waste Spectrum 3							Waste Spectrum 4							Waste Spectrum 5						
Name	No.	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10
N-MWABLIQ	50	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	3	0	0
N-MWSOLIQ	51	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
N-MWWASTE	52	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	0	0	0	10	0	1	1	3	0	0
N-TRIPLAT	53	32	0	1	0	0	0	0	32	0	1	0	0	0	0	32	0	1	0	0	0	0	32	0	1	0	0	0	0	32	0	1	0	3	0	0
N-TRITGAS	54	01	0	1	0	0	0	0	01	0	1	0	0	0	0	01	0	1	0	0	0	0	01	0	1	0	0	0	0	01	0	1	0	3	0	0
N-TRISCNT	55	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	31	3	4	1	2	0	0	00	0	1	1	3	0	0
N-TRILIQD	56	01	0	1	1	0	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	31	3	4	1	2	0	0	01	0	1	1	3	0	0
N-TRITRSH	57	10	0	1	0	0	0	0	10	0	1	0	0	0	0	10	0	1	0	0	0	0	10	0	1	0	0	0	0	10	0	1	0	3	0	0
N-TRIFOIL	58	32	0	1	0	0	0	0	32	0	1	0	0	0	0	32	0	1	0	0	0	0	32	0	1	0	0	0	0	32	0	1	0	3	0	0
N-HIGHACT	59	33	3	1	0	0	2	0	33	3	1	0	4	2	0	33	3	1	0	4	2	0	33	3	1	0	4	2	0	33	3	1	0	4	2	0
N-TRITSOR	60	13	1	1	0	0	0	1	33	1	1	0	2	0	1	33	1	1	0	2	0	1	33	1	1	0	2	0	1	23	1	1	0	3	0	1
N-CARBSOR	61	13	1	1	0	0	0	2	33	1	1	0	2	0	2	33	1	1	0	2	0	2	33	1	1	0	2	0	2	23	1	1	0	3	0	2
N-COBLSOR	62	13	1	1	0	0	0	3	33	1	1	0	2	0	3	33	1	1	0	2	0	3	33	1	1	0	2	0	3	23	1	1	0	3	0	3
N-NICKSOR	63	13	1	1	0	0	0	4	33	1	1	0	2	0	4	33	1	1	0	2	0	4	33	1	1	0	2	0	4	23	1	1	0	3	0	4
N-STROSOR	64	13	1	1	0	0	0	5	33	1	1	0	2	0	5	33	1	1	0	2	0	5	33	1	1	0	2	0	5	23	1	1	0	3	0	5
N-CESISOR	65	13	1	1	0	0	0	6	33	1	1	0	2	0	6	33	1	1	0	2	0	6	33	1	1	0	2	0	6	23	1	1	0	3	0	6
N-PLU8SOR	66	13	1	1	0	0	0	7	33	1	1	0	2	0	7	33	1	1	0	2	0	7	33	1	1	0	2	0	7	23	1	1	0	3	0	7
N-PLU9SOR	67	13	1	1	0	0	0	8	33	1	1	0	2	0	8	33	1	1	0	2	0	8	33	1	1	0	2	0	8	23	1	1	0	3	0	8
N-AMERSOR	68	13	1	1	0	0	0	9	33	1	1	0	2	0	9	33	1	1	0	2	0	9	33	1	1	0	2	0	9	23	1	1	0	3	0	9
N-PUBESOR	69	13	1	1	0	0	0	10	33	1	1	0	2	0	10	33	1	1	0	2	0	10	33	1	1	0	2	0	10	23	1	1	0	3	0	10
N-AMBESOR	70	13	1	1	0	0	0	11	33	1	1	0	2	0	11	33	1	1	0	2	0	11	33	1	1	0	2	0	11	23	1	1	0	3	0	11
N-RANEEDS	71	13	1	1	0	0	0	12	33	1	1	0	2	0	12	33	1	1	0	2	0	12	33	1	1	0	2	0	12	23	1	1	0	3	0	12
N-RACELLS	72	13	1	1	0	0	0	13	33	1	1	0	2	0	13	33	1	1	0	2	0	13	33	1	1	0	2	0	13	23	1	1	0	3	0	13
N-RAPLAQU	73	13	1	1	0	0	0	14	33	1	1	0	2	0	14	33	1	1	0	2	0	14	33	1	1	0	2	0	14	23	1	1	0	3	0	14
N-RANPAPP	74	13	1	1	0	0	0	15	33	1	1	0	2	0	15	33	1	1	0	2	0	15	33	1	1	0	2	0	15	23	1	1	0	3	0	15
N-RABESOR	75	13	0	1	0	0	0	16	33	0	1	0	2	0	16	33	0	1	0	2	0	16	33	0	1	0	2	0	16	23	0	1	0	3	0	16
N-RAMISCL	76	13	0	1	0	0	0	17	33	0	1	0	2	0	17	23	0	1	0	2	0	17	33	0	1	0	2	0	17	23	0	1	0	3	0	17
N-RARESIN	77	11	1	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	11	1	1	0	3	0	0
M-NAVYWET	78	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	2	2	0	2	0	0
M-NAVYDRY	79	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	3	0	0
R-HLLWFRP	80	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
R-FUEHARD	81	33	3	1	0	0	3	0	33	3	1	0	4	3	0	33	3	1	0	4	3	0	33	3	1	0	4	3	0	33	3	1	0	4	3	0
R-HULLFRP	82	33	3	1	0	0	4	0	33	3	1	0	4	4	0	33	3	1	0	4	4	0	33	3	1	0	4	4	0	33	3	1	0	4	4	0
R-ILLWFRP	83	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
R-SILIGEL	84	00	0	1	0	0	0	0	00	0	1	0	0	0	0	00	0	1	0	0	0	0	00	0	1	0	0	0	0	00	0	1	0	3	0	0
R-MPCOTRH	85	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
R-MPCOTRL	86	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
R-MPNCTRA	87	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
R-DEGREXT	88	00	0	1	1	0	0	0	00	0	1	1	0	0	0	00	0	1	1	0	0	0	32	3	4	0	2	0	0	00	0	1	1	3	0	0
R-MPRESIN	89	11	1	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	11	1	1	0	3	0	0
R-SBRESIN	90	11	1	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	11	1	1	0	3	0	0
R-SBCOLIQ	91	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
R-SBCOTRA	92	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
R-SBNCTRA	93	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
R-UFFINES	94	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	3	0	0
R-UFK2MUD	95	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	0	0	0	03	0	1	0	3	0	0
R-UFCOTRA	96	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
R-UFNCTRA	97	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
R-PUCOTRA	98	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0

Table B-8. (continued)

Waste Stream Name	No.	Waste Spectrum 1							Waste Spectrum 2							Waste Spectrum 3							Waste Spectrum 4							Waste Spectrum 5						
		I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10	I4	I5	I6	I7	I8	I9	I10
R-PUNCTRA	99	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
R-MOXCOTR	100	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
R-MOXNCTR	101	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
R-MOXSOLN	102	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
P-DECORES	103	33	3	1	0	0	5	0	33	3	1	0	4	5	0	33	3	1	0	4	5	0	33	3	1	0	4	5	0	33	3	1	0	4	5	0
P-DEACINT	104	33	3	1	0	0	6	0	33	3	1	0	4	6	0	33	3	1	0	4	6	0	33	3	1	0	4	6	0	33	3	1	0	4	6	0
P-DEACVES	105	33	3	1	0	0	7	0	33	3	1	0	4	7	0	33	3	1	0	4	7	0	33	3	1	0	4	7	0	33	3	1	0	4	7	0
P-DEACTCO	106	23	1	1	0	0	0	0	23	1	1	0	0	0	0	23	1	1	0	0	0	0	23	1	1	0	0	0	0	23	1	1	0	3	0	0
P-DECONME	107	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
P-DECONCO	108	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
P-DETRASH	109	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
P-DERESIN	110	11	1	1	1	0	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	32	3	4	0	2	0	0	11	1	1	1	3	0	0
P-DEFILCR	111	21	1	1	0	0	0	0	33	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	31	1	1	0	3	0	0
P-DEEVAPB	112	23	1	2	1	1	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	32	3	4	0	2	0	0	23	2	2	0	2	0	0
B-DECORES	113	33	3	1	0	0	8	0	33	3	1	0	4	8	0	33	3	1	0	4	8	0	33	3	1	0	4	8	0	33	3	1	0	4	8	0
B-DEACINT	114	33	3	1	0	0	9	0	33	3	1	0	4	9	0	33	3	1	0	4	9	0	33	3	1	0	4	9	0	33	3	1	0	4	9	0
B-DEACVES	115	33	3	1	0	0	10	0	33	3	1	0	4	10	0	33	3	1	0	4	10	0	33	3	1	0	4	10	0	33	3	1	0	4	10	0
B-DEACTCO	116	23	1	1	0	0	0	0	23	1	1	0	0	0	0	23	1	1	0	0	0	0	23	1	1	0	0	0	0	23	1	1	0	3	0	0
B-DECONME	117	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
B-DECONCO	118	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
B-DETRASH	119	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	32	3	4	0	2	0	0	10	1	1	0	3	0	0
B-DERESIN	120	11	1	1	1	0	0	0	21	2	2	1	2	0	0	31	3	3	1	2	0	0	32	3	4	0	2	0	0	11	1	1	1	3	0	0
B-DEEVAPB	121	23	1	2	1	1	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	32	3	4	0	2	0	0	23	2	2	1	2	0	0
W-THORHLW	122	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	2	2	0	2	0	0
W-PUREHLW	123	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	2	2	0	2	0	0
W-COTRASH	124	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	31	3	4	0	2	0	0	10	1	1	0	3	0	0
W-NCSOLID	125	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	0	0	0	11	1	1	0	3	0	0
W-LLWTFRE	126	11	0	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	11	0	1	0	3	0	0
W-FRSRESN	127	11	0	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	11	0	1	0	3	0	0
W-FRSLIQD	128	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-RTSRESN	129	11	0	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	11	0	1	0	3	0	0
W-LTTRASH	130	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	31	3	4	0	2	0	0	10	1	1	0	3	0	0
W-HTTRASH	131	10	1	1	0	0	0	0	10	1	1	0	0	0	0	10	1	1	0	0	0	0	31	3	4	0	2	0	0	10	1	1	0	3	0	0
W-LTEQUIP	132	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
W-HTEQUIP	133	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
W-PDWLIQD	134	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-VITSUPR	135	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-VITWASH	136	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-VITSCRB	137	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-VITMELT	138	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-VITFRAC	139	23	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	23	1	2	0	2	0	0
W-VITZEOL	140	33	1	2	0	1	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	13	1	2	0	3	0	0
W-DDRACKS	141	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
W-DDLTRUB	142	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
W-DDHTRUB	143	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	0	0	0	12	1	1	0	3	0	0
W-DDLTLQD	144	23	1	2	1	1	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	31	3	4	0	2	0	0	23	1	2	1	2	0	0
W-DDHTLQD	145	23	1	2	1	1	0	0	23	2	2	1	2	0	0	31	3	3	1	2	0	0	31	3	4	0	2	0	0	23	1	2	1	2	0	0
W-DDRESIN	146	11	1	1	0	0	0	0	23	2	2	0	2	0	0	31	3	3	0	2	0	0	31	3	4	0	2	0	0	11	1	1	0	3	0	0
L-SPENTFU	147	33	3	1	0	0	0	0	33	3	1	0	4	0	0	33	3	1	0	4	0	0	33	3	1	0	4	0	0	33	3	1	0	4	0	0
L-FUEHARD	148	33	3	1	0	0	11	0	33	3	1	0	4	11	0	33	3	1	0	4	11	0	33	3	1	0	4	11	0	33	3	1	0	4	11	0

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Table B-9. Definitions of Subindices for Waste Processing Parameters Index (I1)

Symbol	Meaning	Value		Optional Values
IPK	Packaging Index	0	<	Record number of file containing package distribution
IPR	Processing Index	0	=	No Volume Reduction
		1	=	Regular Compaction
		2	=	Improved Compaction
		3	=	Hydraulic Press
		4	=	Evaporation
		5	=	Pathological Incineration
		6	=	Small Calciner
		7	=	Large Calciner
ISL	Stabilization Index	0	=	No Solidification
		1	=	Solidification Scenario A
		2	=	Solidification Scenario B
		3	=	Solidification Scenario C
		4	=	High Integrity Container
		5	=	Stabilize by another means
ILO	Processing Location	0	=	No Processing
		1	=	Processing at the Generator
		2	=	Processing at the Disposal Site
IEN	Processing Environment	0	=	No Incineration
		1	=	Urban Environment
		2	=	Rural Environment
IRE	Processing Radiation Environment	1	=	High Facility Background Radiation
		2	=	Low Facility Background Radiation

The first index, IPK, is unique in that it can be more than a single digit, and is used to denote the packaging characteristics of the waste stream as it is shipped. It is used to access a set of seven numbers that denote for a given waste stream what percentage is shipped in each of seven modes: unpackaged, in large boxes, in small boxes, in drums, in small liners, in large liners, and in disposable concrete overpacks. The set of seven numbers is used to calculate exposures and costs associated with waste transport and disposal operations. Table B-10 contains the reference package distributions assumed in the work. The particular IPK value assigned for each waste stream in each waste spectrum was chosen based on information presented in Section B.2. Additional distributions may be added at the user's option.

The second index (IPR) denotes the volume reduction process (if any) utilized for the waste stream. An index value of (0) implies no volume reduction. Index values of (1), (2), and (3) indicate routine compaction, improved compaction, and hydraulic press compaction, respectively. An index value of (4) indicates evaporation, and index values of (5), (6), and (7) indicate incineration using a pathological incinerator, fluidized-bed calcination at a small facility, and fluidized-bed calcination at a large facility, respectively.

The third index (ISL) denotes the stabilization processes (if any) applied to the waste stream. An index value of (0) indicates an unsolidified waste form. Index values of (1), (2), and (3) indicate use of solidification scenarios A,

Table B-10. Reference Package Distributions (Percent)

Packaging Index-IPK	Reference Packages LB*	SB	DD	SL	LL
1	0	0	69	15	16
2	23	8	69	0	0
3	0	3	97	0	0
4	0	0	0	0	0
5	100	0	0	0	0
6	0	100	0	0	0
7	0	0	100	0	0
8	0	0	0	100	0
9	0	0	0	0	100
10	0	0	0	0	0
11	90	0	10	0	0
12	0	0	15	15	70
13	28	16	56	0	0
14	10	5	85	0	0
15	70	30	0	0	0

*Reference packages and waste volumes per package are as follows:
LB = Large wooden boxes (128 ft^3);
SB = small boxes (16 ft^3);
DD = 55-gallon drums (7.5 ft^3);
SL = small liners (50 ft^3); and
LL = large liners (170 ft^3).

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B, and C, respectively. An index value of (4) indicates placement into an HIC, while an index value of (5) indicates stabilization by another means.*

The fourth index (ILO) denotes whether the processing (if any) takes place at the waste generator or at a centralized processing facility. An index value of (0) indicates no processing; an index value of (1) indicates processing by the waste generator; and an index value of (2) indicates processing at the centralized processing facility. "Processing" is taken to include compaction (IPR=1-3), improved evaporation (IPR=4), incineration (IPR=5-7); or solidification (ISC=1-3).

The fifth index (IBN) indicates the environment of the location at which the processing is assumed to occur. An index value of (0) indicates no processing; an index value of (1) indicates an urban environment; and an index value of (2) indicates a rural environment.

The last index (IRE) is used to determine occupational exposures due to waste processing activities. An index value of (1) indicates a radiation environment associated with a facility assumed to have a relatively high background radiation level. An index value of 2 or higher indicates other facilities.

The values assigned for these indices for all the waste streams and the waste spectra considered in this report are presented in Table B-11.

B.3.2.3 Volume Reduction and Volume Increase Factors

These factors were previously introduced in Chapter B.2. The volume reduction factor (VRF) is the ratio of the volume of the untreated input waste to the volume of the treated waste product. It is used in quantifying the effects of the volume reduction processes discussed in Section B.2.1. The volume increase factor (VIF) is defined as the ratio of the volume of the product waste stream to the volume of the input waste stream. It is used in quantifying the effects of the volume increase processes discussed in Section B.2.2.

The volume reduction and volume increase factors associated with each of the waste streams for each of the five waste spectra considered in this appendix are presented in Table B-11.

B.3.2.4 Waste Densities

Waste densities are assumed to vary depending upon the particular waste processing option adopted, and so also vary depending upon waste spectra. Waste densities are estimated based on information contained in Appendix A and also from a number of other sources (Refs. 4, 10, 18-22).

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*Similar to the final Part 61 EIS, costs for "stabilization by another means" are assumed to be \$450 and 10 man-hours per m³ of waste.

Table B-11. Processing Indices and Other Parameters for Waste Spectra 1-5

Waste Stream		Waste Spectrum 1				Waste Spectrum 2				Waste Spectrum 3				Waste Spectrum 4				Waste Spectrum 5			
Name	No.	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens
P-IXRESIN	1	1200001	1.00	1.00	.9	1201101	1.00	1.40	1.7	1202101	1.00	2.00	1.2	763121	18.	2.00	1.2	1204001	1.00	1.00	.9
P-CONCLIQ	2	1201101	1.00	1.40	1.7	1201101	1.00	1.40	1.7	1242101	6.00	2.00	1.2	1263121	8.00	2.00	1.2	1201101	1.00	1.40	1.7
P-FSLUDGE	3	1200001	1.00	1.00	.9	1201101	1.00	1.40	1.7	1202101	1.00	2.00	1.2	763121	5.00	2.00	1.2	1204001	1.00	1.00	.9
P-FCARTRG	4	700001	1.00	1.00	1.3	701101	1.00	1.00	1.7	702101	1.00	1.00	1.2	703101	1.00	1.00	1.2	704101	1.00	1.00	1.3
B-IXRESIN	5	1200001	1.00	1.00	.9	1201101	1.00	1.40	1.7	1202101	1.00	2.00	1.2	763121	18.	2.00	1.2	1204001	1.00	1.00	.9
B-CONCLIQ	6	1201101	1.00	1.40	1.7	1201101	1.00	1.40	1.7	1242101	2.40	2.00	1.2	1263121	6.40	2.00	1.2	1201101	1.00	1.40	1.7
B-FSLUDGE	7	1200001	1.00	1.00	.9	1201101	1.00	1.40	1.7	1202101	1.00	2.00	1.2	763121	5.00	2.00	1.2	1204001	1.00	1.00	.9
P-COTRASH	8	1410101	3.00	1.00	.4	1410101	3.00	1.00	.4	1420101	6.00	1.00	.8	763121	80.	2.00	1.2	1414101	3.00	1.00	.4
P-NCTRASH	9	1500001	1.00	1.00	.4	1500001	1.00	1.00	.4	1500001	1.00	1.00	.4	1530201	6.00	1.00	2.4	1504001	1.00	1.00	.4
B-COTRASH	10	1310101	2.00	1.00	.3	1310101	2.00	1.00	.3	1320101	6.00	1.00	.8	763121	80.	2.00	1.2	1314101	2.00	1.00	.3
B-NCTRASH	11	1500001	1.00	1.00	.4	1500001	1.00	1.00	.4	1500001	1.00	1.00	.4	1530201	6.00	1.00	2.4	1504001	1.00	1.00	.4
L-NFRCOMP	12	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
L-DECONRS	13	1200001	1.00	1.00	.9	1201101	1.00	1.40	1.7	1202101	1.00	2.00	1.2	763121	18.	2.00	1.2	1204001	1.00	1.00	.9
F-PROCESS	14	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	704002	1.00	1.00	1.0
F-COTRASH	15	710102	1.50	1.00	.2	710102	1.50	1.00	.2	720102	6.00	1.00	.8	763122	40.	2.00	1.2	714102	1.50	1.00	.2
F-NCTRASH	16	500002	1.00	1.00	.4	500002	1.00	1.00	.4	500002	1.00	1.00	.4	530202	6.00	1.00	2.4	504002	1.00	1.00	.4
U-PROCESS	17	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	704002	1.00	1.00	1.0
L-PUDECON	18	500002	1.00	1.00	1.6	500002	1.00	1.00	1.6	500002	1.00	1.00	1.6	500002	1.00	1.00	1.6	504002	1.00	1.00	1.6
L-BURNUPS	19	701101	1.00	1.00	1.7	701101	1.00	1.00	1.7	702101	1.00	1.43	1.4	703101	1.00	1.43	1.4	704001	1.00	1.00	1.7
I-COTRASH	20	710102	2.00	1.00	.3	710102	2.00	1.00	.3	720102	6.00	1.00	.8	753112	20.	2.00	1.2	714102	2.00	1.00	.3
I+COTRASH	21	700002	1.00	1.00	.1	700002	1.00	1.00	.1	720202	6.00	1.00	.8	773222	80.	2.00	1.2	704002	1.00	1.00	.1
I-ABSLIQD	22	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	753112	100.	2.00	1.2	704002	1.00	3.00	1.0
I+ABSLIQD	23	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	704002	1.00	3.00	1.0
I-LIQSCVL	24	700002	1.00	3.00	.9	700002	1.00	3.00	.9	710102	1.28	3.00	.9	753112	4.52	2.00	1.2	704002	1.00	3.00	.9
I+LIQSCVL	25	700002	1.00	3.00	.9	700002	1.00	3.00	.9	710202	1.28	3.00	.9	710202	1.28	3.00	.9	704002	1.00	3.00	.9
I-BIOWAST	26	700002	1.00	1.92	1.1	700002	1.00	1.92	1.1	700002	1.00	1.92	1.1	753112	15.	2.00	1.2	704002	1.00	1.92	1.1
I+BIOWAST	27	700002	1.00	1.92	1.1	700002	1.00	1.92	1.1	700002	1.00	1.92	1.1	700002	1.00	1.92	1.1	704002	1.00	1.92	1.1
N-SSTRASH	28	710102	1.50	1.00	.2	710102	1.50	1.00	.2	720102	5.00	1.00	.6	753112	10.	2.00	1.2	714102	1.50	1.00	.2
N+SSTRASH	29	700002	1.00	1.00	.1	700002	1.00	1.00	.1	720202	5.00	1.00	.6	773222	40.	2.00	1.2	704002	1.00	1.00	.1
N-SSWASTE	30	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	704002	1.00	1.00	1.0
N-LOTRASH	31	710102	2.00	1.00	.3	710102	2.00	1.00	.3	720102	6.00	1.00	.8	753112	20.	2.00	1.2	714102	2.00	1.00	.3
N+LOTRASH	32	700002	1.00	1.00	.1	700002	1.00	1.00	.1	720202	6.00	1.00	.8	773222	80.	2.00	1.2	704002	1.00	1.00	.1
N-LOWASTE	33	700002	1.00	1.00	.5	700002	1.00	1.00	.5	700002	1.00	1.00	.5	700002	1.00	1.00	.5	704002	1.00	1.00	.5
N-ISOPROD	34	701001	1.00	1.00	1.7	701101	1.00	1.00	1.7	702101	1.00	1.43	1.2	703101	1.00	1.43	1.2	704001	1.00	1.00	1.7
N-ISOTRSH	35	500001	1.00	1.00	.6	500001	1.00	1.00	.6	500001	1.00	1.00	.6	530201	6.00	1.00	2.4	504001	1.00	1.00	.6
N-SORMFG1	36	700002	1.00	1.00	2.0	700002	1.00	1.00	2.0	700002	1.00	1.00	2.0	700002	1.00	1.00	2.0	704002	1.00	1.00	2.0
N-SORMFG2	37	1100001	1.00	1.00	.4	1100001	1.00	1.00	.4	1100001	1.00	1.00	.4	1100001	1.00	1.00	.4	1104001	1.00	1.00	.4
N-SORMFG3	38	700001	1.00	2.00	.4	700001	1.00	2.00	.4	700001	1.00	2.00	.4	700001	1.00	2.00	.4	704001	1.00	2.00	.4
N-SORMFG4	39	500002	1.00	1.00	.4	500002	1.00	1.00	.4	500002	1.00	1.00	.4	530202	6.00	1.00	2.4	504002	1.00	1.00	.4
N-NECOTRA	40	710102	2.00	1.00	.3	710102	2.00	1.00	.3	720102	6.00	1.00	.8	753112	20.	2.00	1.2	714102	2.00	1.00	.3
N-NEABLIQ	41	700002	1.00	4.10	.9	700002	1.00	4.10	.9	700002	1.00	4.10	.9	700002	1.00	4.10	.9	704002	1.00	4.10	.9
N-NESOLIQ	42	701102	1.00	1.40	1.7	701102	1.00	1.40	1.7	702102	1.00	2.00	1.2	703102	1.00	2.00	1.2	701102	1.00	1.40	1.7
N-NEVIALS	43	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	700002	1.00	3.00	1.0	704002	1.00	3.00	1.0
N-NENCGLS	44	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	704002	1.00	1.00	1.0
N-NEWOTAL	45	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	704002	1.00	1.00	1.0
N-NETRGAS	46	704002	1.00	5.35	.0	704002	1.00	5.35	.0	704002	1.00	5.35	.0	704002	1.00	5.35	.0	704002	1.00	5.35	.0
N-NETRILI	47	704002	1.00	5.35	.9	704002	1.00	5.35	.9	704002	1.00	5.35	.9	704002	1.00	5.35	.9	704002	1.00	5.35	.9
N-NECARLI	48	704002	1.00	5.35	.9	704002	1.00	5.35	.9	704002	1.00	5.35	.9	704002	1.00	5.35	.9	704002	1.00	5.35	.9
N-MWTRASH	49	700002	1.00	1.00	.4	700002	1.00	1.00	.4	700002	1.00	1.00	.4	700002	1.00	1.00	.4	704002	1.00	1.00	.4

Table B-11. (continued)

Waste Stream		Waste Spectrum 1				Waste Spectrum 2				Waste Spectrum 3				Waste Spectrum 4				Waste Spectrum 5			
Name	No.	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens
N-MWABLIQ	50	700002	1.00	5.50	1.0	700002	1.00	5.50	1.0	700002	1.00	5.50	1.0	700002	1.00	5.50	1.0	704002	1.00	5.50	1.0
N-MWSOLIQ	51	701102	1.00	1.40	1.7	701102	1.00	1.40	1.7	702102	1.00	2.00	1.2	703102	1.00	2.00	1.2	701102	1.00	1.40	1.7
N-MWWASTE	52	700002	1.00	1.00	.4	700002	1.00	1.00	.4	700002	1.00	1.00	.4	700002	1.00	1.00	.4	704002	1.00	1.00	.4
N-TRIPLAT	53	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	700002	1.00	1.00	1.0	704002	1.00	1.00	1.0
N-TRITGAS	54	700002	1.00	1.00	.0	700002	1.00	1.00	.0	700002	1.00	1.00	.0	700002	1.00	1.00	.0	704002	1.00	1.00	.0
N-TRISCNT	55	700002	1.00	3.00	.9	700002	1.00	3.00	.9	710102	1.28	3.00	.9	753112	4.52	2.00	1.2	704002	1.00	3.00	.9
N-TRILIQD	56	700002	1.00	3.00	1.0	701102	1.00	1.40	1.7	702102	1.00	2.00	1.2	753112	100.	2.00	1.2	704002	1.00	3.00	1.0
N-TRITRSH	57	700002	1.00	1.00	.1	700002	1.00	1.00	.1	700002	1.00	1.00	.1	730202	6.00	1.00	.8	704002	1.00	1.00	.1
N-TRIFOIL	58	700002	1.00	1.00	.4	700002	1.00	1.00	.4	700002	1.00	1.00	.4	730202	6.00	1.00	2.4	704002	1.00	1.00	.4
N-HIGHACT	59	700002	1.00	1.00	7.8	705102	1.00	1.00	7.8	705102	1.00	1.00	7.8	705102	1.00	1.00	7.8	705102	1.00	1.00	7.8
N-TRITSOR	60	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-CARBSOR	61	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-COBLSOR	62	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-NICKSOR	63	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-STROSOR	64	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-CESISOR	65	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-PLU8SOR	66	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-PLU9SOR	67	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-AMERSOR	68	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-PUBESOR	69	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-AMBESOR	70	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RANEEDS	71	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RACELLS	72	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RAPLAQU	73	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RANPAPP	74	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RABESOR	75	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RAMISCL	76	700002	1.00	1.00	.4	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	701102	1.00	1.00	1.7	704002	1.00	1.00	1.0
N-RARESIN	77	800002	1.00	1.00	.9	801102	1.00	1.40	1.7	802102	1.00	2.00	1.2	873112	18.	2.00	1.2	804002	1.00	1.00	.9
M-NAVYWET	78	701101	1.00	1.00	.4	701101	1.00	1.00	.4	702101	1.00	1.43	.4	763111	8.00	2.00	2.4	701101	1.00	1.00	.4
M-NAVYDRY	79	700001	1.00	1.00	1.7	700001	1.00	1.00	1.7	700001	1.00	1.00	1.2	730201	6.00	1.00	1.2	704001	1.00	1.00	1.7
R-HLLWFRP	80	1201101	1.00	1.40	1.7	1201101	1.00	1.40	1.7	1242101	6.00	2.00	1.2	1263121	8.00	2.00	1.2	1201101	1.00	1.40	1.7
R-FUEHARD	81	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
R-HULLFRP	82	800001	1.00	1.00	1.0	805101	1.00	1.00	1.0	805101	1.00	1.00	1.0	805101	1.00	1.00	1.0	805101	1.00	1.00	1.0
R-ILLWFRP	83	801101	1.00	1.40	1.7	801101	1.00	1.40	1.7	842101	6.00	2.00	1.2	763121	8.00	2.00	1.2	801101	1.00	1.40	1.7
R-SILIGEL	84	700001	1.00	1.00	.8	700001	1.00	1.00	.8	700001	1.00	1.00	.8	700001	1.00	1.00	.8	704001	1.00	1.00	.8
R-MPCOTRH	85	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4
R-MPCOTRL	86	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4
R-MPNCTRA	87	500001	1.00	1.00	.4	500001	1.00	1.00	.4	500001	1.00	1.00	.4	530201	6.00	1.00	2.4	504001	1.00	1.00	.4
R-DEGREXT	88	700101	1.00	3.00	.8	700101	1.00	3.00	.8	700101	1.00	3.00	.8	763121	4.50	2.00	1.2	704101	1.00	3.00	.8
R-MPRESIN	89	900001	1.00	1.00	.9	901101	1.00	1.40	1.7	902101	1.00	2.00	1.2	763121	18.	2.00	1.2	904001	1.00	1.00	.9
R-SBRESIN	90	900001	1.00	1.00	.9	901101	1.00	1.40	1.7	902101	1.00	2.00	1.2	763121	18.	2.00	1.2	904001	1.00	1.00	.9
R-SBCOLIQ	91	901101	1.00	1.40	1.7	901101	1.00	1.40	1.7	942101	6.00	2.00	1.2	763121	8.00	2.00	1.2	901101	1.00	1.40	1.7
R-SBCOTRA	92	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4
R-SBNCTRA	93	500001	1.00	1.00	.4	500001	1.00	1.00	.4	500001	1.00	1.00	2.4	530201	6.00	1.00	2.4	504001	1.00	1.00	2.4
R-UFFINES	94	700001	1.00	1.00	1.5	700001	1.00	1.00	1.5	700001	1.00	1.00	1.5	700001	1.00	1.00	1.5	704001	1.00	1.00	1.5
R-UFK2MUD	95	700001	1.00	1.00	1.2	700001	1.00	1.00	1.2	700001	1.00	1.00	1.2	700001	1.00	1.00	1.2	704001	1.00	1.00	1.2
R-UFCOTRA	96	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4
R-UFNCTRA	97	500001	1.00	1.00	.4	500001	1.00	1.00	.4	500001	1.00	1.00	.4	530201	6.00	1.00	2.4	504001	1.00	1.00	.4
R-PUCOTRA	98	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4

Table B-11. (continued)

Waste Stream		Waste Spectrum 1				Waste Spectrum 2				Waste Spectrum 3				Waste Spectrum 4				Waste Spectrum 5			
Name	No.	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens	I1	VRF	VIF	Dens
R-PUNCTRA	99	500001	1.00	1.00	.4	500001	1.00	1.00	.4	500001	1.00	1.00	.4	530201	6.00	1.00	2.4	504001	1.00	1.00	.4
R-MOXCOTR	100	710102	3.00	1.00	.4	710102	3.00	1.00	.4	720102	6.00	1.00	.8	763122	80.	2.00	1.2	714102	3.00	1.00	.4
R-MOXNCTR	101	500002	1.00	1.00	.4	500002	1.00	1.00	.4	500002	1.00	1.00	.4	530202	6.00	1.00	2.4	504002	1.00	1.00	.4
R-MOXSOLN	102	701102	1.00	1.40	1.7	701102	1.00	1.40	1.7	742102	6.00	2.00	1.2	763122	8.00	2.00	1.2	701102	1.00	1.40	1.7
P-DECORES	103	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
P-DEACINT	104	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
P-DEACVES	105	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
P-DEACTCO	106	500001	1.00	1.00	4.5	500001	1.00	1.00	4.5	500001	1.00	1.00	4.5	500001	1.00	1.00	4.5	504001	1.00	1.00	4.5
P-DECONME	107	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
P-DECONCO	108	500001	1.00	1.00	3.0	500001	1.00	1.00	3.0	500001	1.00	1.00	3.0	500001	1.00	1.00	3.0	504001	1.00	1.00	3.0
P-DETRASH	109	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4
P-DERESIN	110	900001	1.00	1.00	.9	901101	1.00	1.40	1.7	902101	1.00	2.00	1.2	763121	18.	2.00	1.2	904001	1.00	1.00	.9
P-DEFILCR	111	700001	1.00	1.00	1.3	701101	1.00	1.00	1.7	702101	1.00	1.00	1.2	703101	1.00	1.00	1.2	704001	1.00	1.00	1.3
P-DEEVAPB	112	901101	1.00	1.40	1.7	901101	1.00	1.40	1.7	942101	6.00	2.00	1.2	763121	8.00	2.00	1.2	901101	1.00	1.40	1.7
B-DECORES	113	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
B-DEACINT	114	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
B-DEACVES	115	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8
B-DEACTCO	116	500001	1.00	1.00	4.5	500001	1.00	1.00	4.5	500001	1.00	1.00	4.5	500001	1.00	1.00	4.5	504001	1.00	1.00	4.5
B-DECONME	117	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
B-DECONCO	118	500001	1.00	1.00	3.0	500001	1.00	1.00	3.0	500001	1.00	1.00	3.0	500001	1.00	1.00	3.0	504001	1.00	1.00	3.0
B-DETRASH	119	710101	3.00	1.00	.4	710101	3.00	1.00	.4	720101	6.00	1.00	.8	763121	80.	2.00	1.2	714101	3.00	1.00	.4
B-DERESIN	120	900001	1.00	1.00	.9	901101	1.00	1.40	1.7	902101	1.00	2.00	1.2	763121	18.	2.00	1.2	904001	1.00	1.00	.9
B-DEEVAPB	121	901101	1.00	1.40	1.7	901101	1.00	1.40	1.7	942101	2.40	2.00	1.2	963121	6.40	2.00	1.2	901101	1.00	1.40	1.7
W-THORHLW	122	1201101	1.00	1.40	1.7	1201101	1.00	1.40	1.7	1242101	6.00	2.00	1.2	763121	8.00	2.00	1.2	1201101	1.00	1.40	1.7
W-PUREHLW	123	1201101	1.00	1.40	1.7	1201101	1.00	1.40	1.7	1242101	6.00	2.00	1.2	763121	8.00	2.00	1.2	1201101	1.00	1.40	1.7
W-COTRASH	124	510101	3.00	1.00	.4	510101	3.00	1.00	.4	520101	6.00	1.00	.8	763121	80.	2.00	1.2	514101	3.00	1.00	.4
W-NCSOLID	125	500001	1.00	1.00	.4	500001	1.00	1.00	.4	500001	1.00	1.00	.4	530101	6.00	1.00	2.4	504001	1.00	1.00	.4
W-LLWTFRE	126	700001	1.00	1.00	.9	701101	1.00	1.40	1.7	702101	1.00	2.00	1.2	763121	5.00	2.00	1.2	704001	1.00	1.00	.9
W-FRSRESN	127	800001	1.00	1.00	.9	801101	1.00	1.40	1.7	802101	1.00	2.00	1.2	863121	5.00	2.00	1.2	804001	1.00	1.00	.9
W-FRSLIQD	128	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-RTSRESN	129	700001	1.00	1.00	.9	701101	1.00	1.40	1.7	702101	1.00	2.00	1.2	763121	5.00	2.00	1.2	704001	1.00	1.00	.9
W-LTTRASH	130	510101	3.00	1.00	.4	510101	3.00	1.00	.4	520101	6.00	1.00	.8	763121	80.	2.00	1.2	514101	3.00	1.00	.4
W-HTTRASH	131	510101	3.00	1.00	.4	510101	3.00	1.00	.4	520101	6.00	1.00	.8	763121	80.	2.00	1.2	514101	3.00	1.00	.4
W-LTEQUIP	132	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
W-HTEQUIP	133	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
W-PDWLIQD	134	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-VITSUPR	135	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-VITWASH	136	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-VITSCRB	137	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-VITMELT	138	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-VITFRAC	139	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-VITZEOL	140	700001	1.00	1.00	1.0	701101	1.00	1.40	1.7	702101	1.00	2.00	1.2	763121	8.00	2.00	1.2	704001	1.00	1.00	1.0
W-DDRACKS	141	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
W-DDLTRUB	142	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
W-DDHTRUB	143	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	500001	1.00	1.00	2.0	504001	1.00	1.00	2.0
W-DDLTLQD	144	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-DDHTLQD	145	701101	1.00	1.40	1.7	701101	1.00	1.40	1.7	742101	6.00	2.00	1.2	763121	8.00	2.00	1.2	701101	1.00	1.40	1.7
W-DDRESIN	146	700001	1.00	1.00	.9	701101	1.00	1.40	1.7	702101	1.00	2.00	1.2	763121	18.	2.00	1.2	704001	1.00	1.00	.9
L-SPENTFU	147	800001	2.00	1.00	8.0	805101	2.00	1.00	8.0	805101	2.00	1.00	8.0	805101	2.00	1.00	8.0	805101	2.00	1.00	8.0
L-FUEHARD	148	800001	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8	805101	1.00	1.00	7.8

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The densities (not specific gravities) are used as an input to the waste classification assessments in the computer codes. Given this, some judgement is required on how to assign the densities. In general, densities are estimated in terms of the final waste form exclusive of waste container. An exception is absorbed aqueous and scintillation liquids, in which the densities are assigned based on the liquids prior to absorbtion. This is in keeping with the Waste Classification Technical Position (Ref. 23). In any case, the waste densities are only important for classifying transuranic radioisotopes.

The waste densities are to be used as a general guide as an aid to decision-making, and should not be used as a substitute for real data on specific wastes generated by specific licensees. A number of typical densities are listed in Table B-12.

Densities for LWR compactible trash are estimated based on an average uncompactible waste density of 0.13 g/cc (8 lb/ft^3--Ref. 4). The density of the final waste form depends on the volume reduction applied. A similar approach is taken for other fuel cycle and non fuel cycle waste streams.

Assumed waste densities for each waste stream and each waste spectra are listed in Table B-11.

Table B-12. Typical Densities of Different Materials

Waste Form	Density (g/cm^3)
Waste solidified in cement	1.7
Waste solidified in vinyl ester styrene	1.2
PWR filter cartridges, unsolidified	1.3
Dewatered ion exchange resins	0.9
Dewatered filter sludge	0.9
Uncompacted compressible trash	0.13
PWR compacted trash (VR=3)	0.4
BWR compacted trash (VR=2)	0.3
Aqueous liquids	1.0
Scintillation liquids	0.9
Structural concrete	3.0
High density concrete	4.5
Rolled steel	7.85
Lead	11.4
Wood	0.4 - 0.7
Demolition material, mixed, noncombustible	1.4
Broken pavement or sidewalk	1.5
Dirt, sand or gravel (uncompacted)	1.4
Biological waste	1.1
Air	0.0013

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APPENDIX C

ALTERNATIVE DISPOSAL METHODS AND SITE ENVIRONMENTS

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APPENDIX C. Alternative Disposal Methods and Site Environments

C.1 INTRODUCTION

This appendix provides a description of the alternative disposal methods that will be considered in this report, along with the assumptions and methodology that will be used to assess disposal costs. Such costs include preoperational, operational, and postoperational activities. A present value analysis is used to determine charges that a disposal facility operator would assess to recover capital and operational expenses, as well as surcharges to cover closure and institutional control costs. The appendix also contains a description of the environmental characteristics of the four reference disposal facility sites assumed in this report, as well as values for environmental parameters used in the calculational methodology.

In the appendix, a description of a large disposal facility is first provided (Chapter C.2) which is used as a reference against which the disposal alternatives are addressed. This reference disposal facility is assumed to be located in a humid environment and is essentially the same facility that is described in the draft environmental impact statement (EIS) (Ref. 1) on the regulation 10 CFR Part 61. Much of the description of this facility is in fact taken from Appendix E of the draft EIS. This description is intended to introduce basic concepts such as the activities carried out over the facility lifetime, as well as the basic appearance of the disposal facility. Following this, a brief description of the disposal alternatives is presented in Chapter C.3.

Following this, a cost analysis is presented in Chapters C.4 and C.5. First, capital, operational, and postoperational (closure, surveillance, and institutional control periods) costs are detailed for the disposal various technologies assuming that a facility disposes from 2,000 to 50,000 m^3 of waste over an arbitrary (35 year maximum) operational period. Up to six different disposal methods may be simultaneously used on a given site (Chapter C.4). The time value of money is next considered in Chapter 6.0. This chapter includes a present value analysis methodology which is updated from the present value analysis outlined in the final Part 61 EIS (Ref. 2).

The final chapter on alternative disposal site environments (Chapter C.7) is based upon Appendix C of reference 3.

C.2 BASIC SITING, DESIGN, OPERATION, AND CLOSURE CONCEPTS

This chapter provides a description of the reference disposal facility as well as a description of the basic concepts associated with siting, designing, operating, and closing a disposal facility. This description is oriented toward the reference disposal facility, but the basic concepts are applicable to any disposal method.

In developing the reference disposal facility, the authors were influenced by past history and experience at shallow land burial disposal facilities, and by the desire to emphasize potential long-term costs and radiological impacts in this report. For example, a great deal of experience has been gained over the years regarding handling and disposal of radioactive material. Safe working procedures have been recognized and developed such as the need to maintain strict control of potential site contamination to help minimize personnel exposures and potential offsite radioactivity releases during site operations. The development and application of good siting criteria and operational radiation safety practices are to a large part an application of common sense. The main focus of this report is the potential long-term costs and impacts of waste disposal.

In this appendix, therefore, the reference disposal facility is developed assuming that an application for a new facility is received, and consideration is given to closure and institutional control issues during the subsequent regulatory application review and licensing process. A number of common sense siting considerations and radiation safety practices are also assumed. However, given these siting considerations and radiation safety practices, no special effort is assumed to be made in the reference facility to ensure long-term site stability. This is used to provide a base case level of long-term costs and radiological impacts against which measures to achieve site stability, to minimize radiological impacts, and to ensure adequate funding can be assessed. These measures (e.g., compaction of backfill material and trench caps, and disposal of more stable waste forms) are generally already being applied today as part of existing disposal site license conditions and the Part 61 regulation. They have not been assumed for the reference facility, however, in order that the costs and impacts of incorporating them can be systematically analyzed.

The reference facility is assumed to be located in a humid environment and is sized to accept a relatively large quantity of waste--i.e., 50,000 m^3 of waste per year over a 20-year operating life, or a total volume of 1 million m^3. This corresponds to approximately one-quarter of the total volume of low-level waste projected to be generated in the United States to the year 2000. Disposal of 1 million m^3 of waste in the reference facility described in this appendix will require up to a few hundred acres of land, which corresponds to an approximate upper bound of the land area of current commercial disposal facilities.

As part of the following discussions, reference is made to interactions between an applicant or licensee and a regulatory (licensing) agency. The applicant submits an application to a licensing agency which is then reviewed in accordance with licensing procedures established by the licensing agency. For the purposes of the appendix, the licensing agency is assumed to be NRC and the licensing procedures described are those of NRC. A state regulatory agency may have different procedures.

C.2.1 Preoperational Activities

C.2.1.1 Site Selection and Characterization

Once the need and desire to operate a waste disposal facility have become established, the potential applicant embarks on a site selection study which is assumed to last 1 to 2 years. The intent of the site selection study is to review and evaluate potential locations for a disposal site through a systematic process, and to gather sufficient data to support a license application. There are a number of methods or procedures by which the site selection study may be carried out. A brief outline of a method assumed to be used by the potential applicant is described below.

For the purposes of this appendix, the potential applicant is assumed to first establish a region of interest within which the potential applicant would propose locating a near-surface disposal facility. Depending upon the particular circumstances, this region of interest may be of a variable size--e.g., encompassing a single state or potentially a multiple-station region. From within this region of interest, the potential applicant selects a number of candidate areas within which perhaps 8 to 12 potential sites may be identified. This list of potential sites is then narrowed down to a slate of alternative candidate sites (3 to 5 sites) from which a most-favored site is eventually selected.

In arriving at a slate of candidate sites, the potential applicant principally uses reconnaissance level information in obtaining needed hydrologic, geologic, demographic, and other data. That is, use is made of such information as relevant scientific literature (e.g., topographic, geologic, water resource, biotic, and demographic maps, as well as aerial photographs), reports of government or private research agencies, consultation with experts, and short-term field investigations, as well as analyses performed using such information. The amount of information collected and the extent of analyses conducted increases as the potential applicant moves from consideration of the region of interest to the slate of candidate sites. An additional consideration, which is assumed to be of importance to a potential applicant, is the availability of a good local road network. Socio-economic factors such as current land use and the availability of labor or local utility services are also important considerations to the potential applicant.

To assist in selecting a most-favored site from the slate of candidate sites, the potential applicant is assumed to drill a small number of subsurface reconnaissance wells at each of the slate of candidate sites. This is to help determine the agreement between the regional hydrologic data base and more specific site conditions. The potential applicant is then assumed to purchase the site most favored among the slate of candidate sites and to initiate the detailed subsurface investigation of the site. From this field investigation, the potential applicant is assumed to prepare such items as detailed boring logs, numerous cross-sections of the site geology, a site topographic map, and a site potentiometric surface map for each aquifer of interest. In addition, the potential applicant is assumed to define the engineering and material properties of the soil units used for disposal, backfill, or trench caps, and to prepare a site drainage drawing. (Also see reference 4.)

It is assumed that a range of 25 to 50 test holes of variable depths is drilled to determine the subsurface conditions of the site. Many of the peripherally located test holes are assumed to be subsequently converted into ground-water monitoring wells. The potential applicant commences preoperational monitoring of the site, which helps to provide the data needed to support the license application as well as the baseline from which the effects of site construction and waste disposal are identified. The preoperational monitoring program includes periodic collection of surface water, ground water, biota, soil, and airborne particulate samples by an appropriate method (e.g., grab, continuous or composite sampling). Ground water levels and stream flows are measured periodically. Site meteorological data--particularly precipitation, wind speed and direction at various heights, temperatures, and soil moisture data--are also measured.

Throughout the site-selection phase, the potential applicant is assumed to have had a series of discussions with state representatives regarding custodianship of the disposal facility, as well as funding mechanisms for long-term care of the facility.

During the final year of the site-selection phase, the investigations performed at the favored and candidate sites are assumed to have sufficiently and favorably progressed so that the potential applicant has reasonable confidence that there are no insurmountable technical or political problems. The potential applicant is then assumed to reach a management decision to proceed with the undertaking, and preparation of a license application is initiated.

C.2.1.2 Preoperational Phase

This phase of the facility life span is assumed to last approximately 2 years and mainly consists of submittal of a license application to NRC and subsequent review by NRC licensing staff. During the review period, the applicant continues with the preoperational environmental monitoring program.

Upon receipt of the application, NRC would review the application for completeness. If the application is incomplete, NRC would notify the applicant of the items needed to complete the application. In addition, upon docketing the application, NRC would notify the governor and the state legislature of the state in which the proposed site is located that a license application had been received. It is expected that this notification would be only a formality since considerable prior contact with state representatives by the applicant is highly probable.

The application would include a safety analysis report and an environmental report pursuant to 10 CFR Part 51. The environmental report would include a detailed description of the proposed action, a statement of its purposes, a description of the environment affected, and a description of the potential effects of the facility on that environment. As part of this, the proposed site (previously termed the most favored site among the slate of candidate sites) is described in detail. The potential environmental consequences and methods to mitigate these consequences are addressed, as well as alternatives to the proposed action, including alternative sites.

Also included with the license application is a preliminary site closure plan. This plan will include a detailed plan for waste emplacement, expected capacity of the site, the planned site contours and drainage systems during operations as well as the final site contours, and delineation of the buffer zone. The closure plan will include: (1) estimated costs of labor, equipment and material for closure and stabilization, and (2) long-term labor and material costs for eventual site surveillance, monitoring, and control by the site owner.

Once the complete application has been received and docketed by NRC, the receipt of the application is announced in the Federal Register in compliance with Part 2 of the Commission's regulations. A press release is also issued. In the Federal Register notice, opportunity is provided for persons with an interest in the proposed action to request a hearing. If such a hearing is to be held, NRC will appoint an Atomic Safety and Licensing Board (ASLB) to review the licensing action. Meanwhile, the application is reviewed by NRC licensing staff and a safety evaluation report is prepared. If the information contained in the application is insufficient to prepare the safety evaluation report and reach a decision, additional information may be requested from the applicant.

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Under existing NRC regulations in 10 CFR 51, issuing a license for a waste disposal facility constitutes a major federal action according to the National Environmental Policy Act of 1969. Accordingly, an environmental impact statement (EIS) is prepared by the NRC staff and a draft published for public comment. Based upon public comment received and perhaps based upon additional information obtained from the applicant, a final EIS is prepared and published. Upon consideration of the final EIS, NRC staff will make decisions regarding the application and, if a license is granted by the Commission, any license conditions that the staff believe are necessary. If a hearing is held, NRC licensing staff would recommend a course of action (i.e., either rejecting the application, or granting a license subject to conditions) to the ASLB. Testimony would be presented and intervenors would be given an opportunity to cross-examine witnesses. The ASLB will review all testimony and would ultimately reach a decision on the application. This decision may be appealed to the Atomic Safety and Licensing Appeal Board, then to the Commission, and finally to the courts. Hearings, including preparation and presentation of testimony and preparation of the hearing record leading to a decision, would last approximately 1 year.

After resolution of any hearings and appeals, the NRC staff may issue a license. Before the license can be issued, however, ownership of the disposal facility site must be transferred to either the state or federal government and an acceptable funding arrangement must be provided.

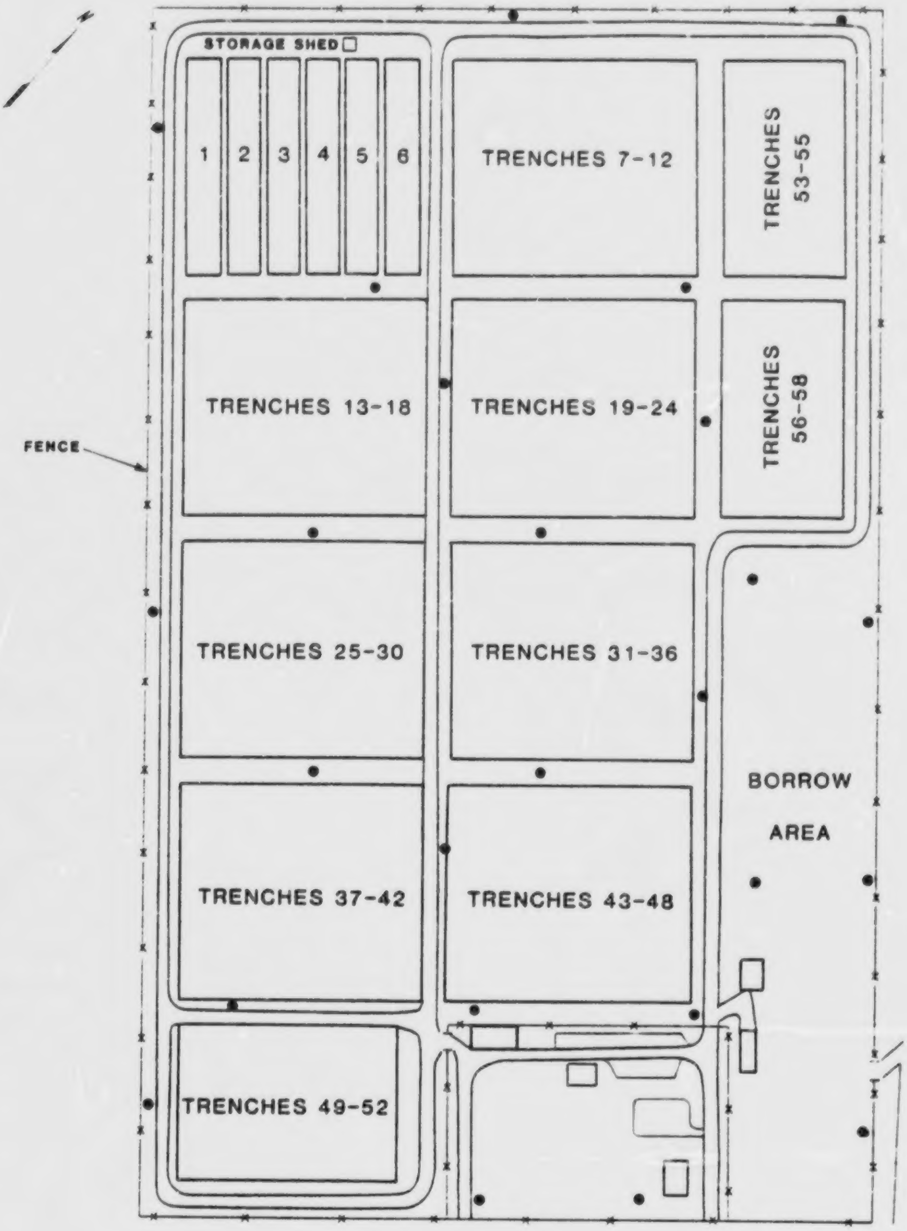
For purposes of this appendix, the facility site is assumed to be transferred by the applicant to the control of an agency of the state in which the site is located. The state is assumed to enter into a lease arrangement with the applicant. The terms and conditions of the lease are assumed to be reviewed on a 10-year basis, with the exception of funding arrangements, which are assumed to be reviewed on a 5-year basis.

Funding arrangements are established as part of the lease, and include specific arrangements to provide funds for: (1) disposal facility closure and stabilization: and (2) long-term site surveillance and control by the site owner (in this case the state). The availability of funding for facility closure and stabilization is assumed to be assured through a surety bond acquired by the applicant. This surety bond would be used by the state to close the site should the applicant default (e.g., go out of business). Otherwise, the applicant would pay for final site closure. Funding for long-term care and surveillance is assumed to be provided by a surcharge on waste received and disposed at the facility. Monies collected from this surcharge are placed into an interest-bearing state account which is dedicated to the long-term care of the facility.

C.2.2 Operational Activities

C.2.2.1 Site Design

A conceptual layout of the reference disposal facility is illustrated in Figures C.1 and C.2. As shown in the figures, the disposal facility may be divided into two basic areas: a "restricted area" and an "administration area." Pursuant to Part 20 of the Commission's regulations, the restricted area is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials. The restricted area


STORAGE SHED
1
2
3
4
5
6
TRENCHES 7-12
TRENCHES 53-55
TRENCHES 13-18
TRENCHES 19-24
TRENCHES 56-58
FENCE
TRENCHES 25-30
TRENCHES 31-36
BORROW AREA
TRENCHES 37-42
TRENCHES 43-48
TRENCHES 49-52

REFERENCE DISPOSAL SITE LAYOUT

Figure C.1

● MONITORING WELLS

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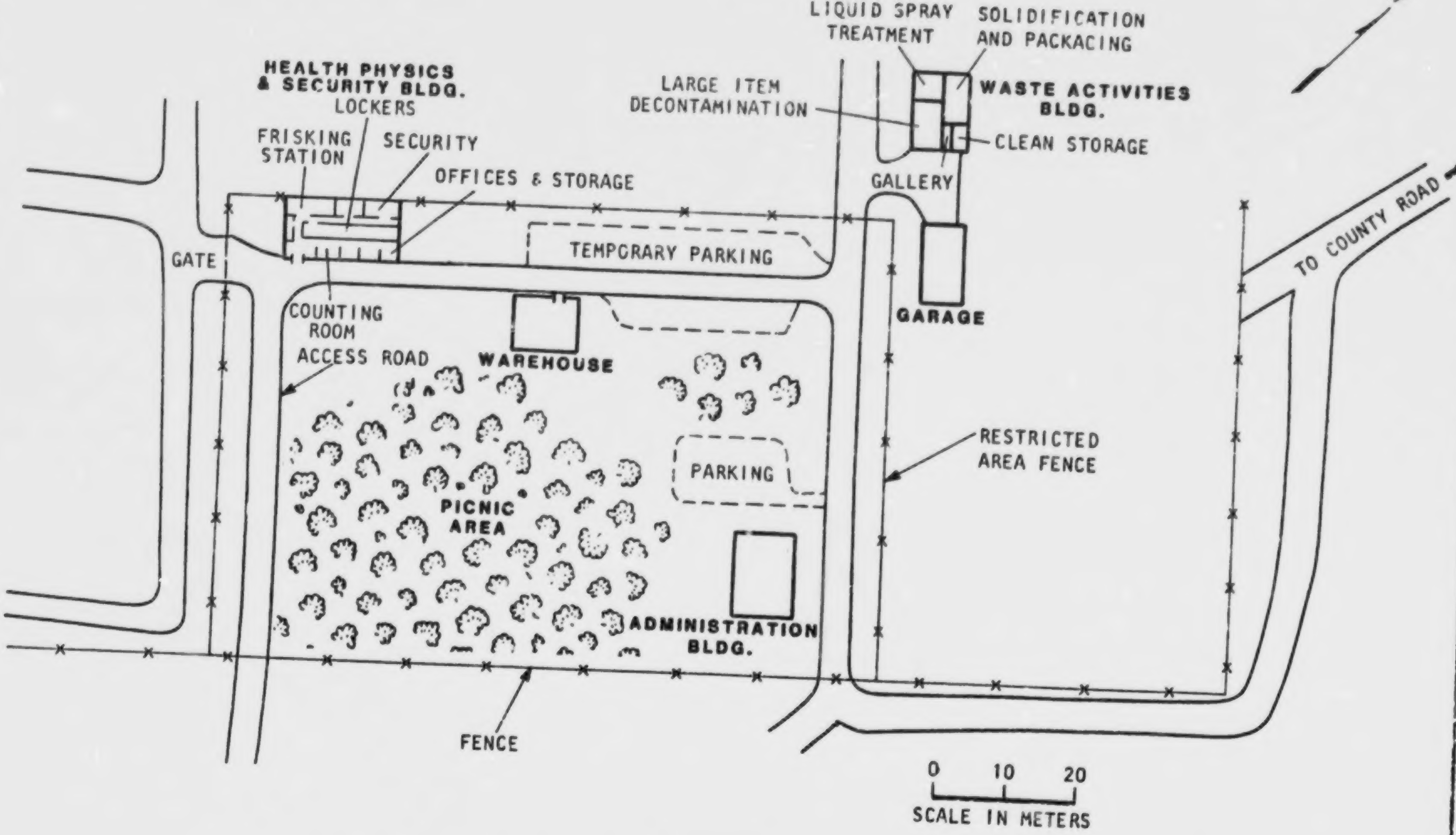

LIQUID SPRAY TREATMENT
SOLIDIFICATION AND PACKACING
HEALTH PHYSICS & SECURITY BLDG.
LOCKERS
LARGE ITEM DECONTAMINATION
WASTE ACTIVITIES BLDG.
FRISKING STATION
SECURITY
CLEAN STORAGE
OFFICES & STORAGE
GALLERY
GATE
TEMPORARY PARKING
TO COUNTY ROAD
COUNTING ROOM
GARAGE
ACCESS ROAD
WAREHOUSE
PICNIC AREA
PARKING
RESTRICTED AREA FENCE
ADMINISTRATION BLDG.
FENCE
0
10
20
SCALE IN METERS

Figure C.2. Reference Disposal Facility Support Facilities

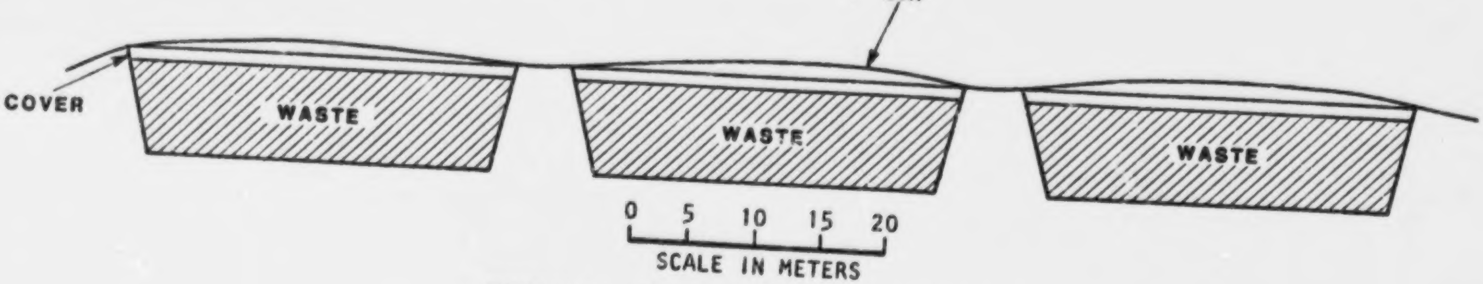
includes a "disposal area," in which disposal of radioactive waste takes place, as well as an "operational area." As shown, the restricted area includes a buffer zone between the disposal trenches and the restricted area fence of about 30 m (100 ft). As shown in Figure C.1., the operational area is located along the southern and eastern sides of the disposal facility and is used as a borrow area, for cask storage, and for other miscellaneous functions. The operational area includes two facilities, a decontamination facility and a garage, which are used to support waste disposal operations. The administration area is located near the eastern corner of the disposal facility and is considered uncontrolled by the licensee for purposes of radiation protection. The administration area includes support facilities plus parking space for employees as well as for incoming waste delivery vehicles.

As is the case at existing disposal facilities, considerably less than the total site acreage purchased by the site operator is assumed to be used for waste disposal. For example, specific areas of a particular disposal site may not be suitable for waste disposal due to geohydrological or topographical reasons--e.g., parts of a particular site might have excessively steep slopes or high water tables. The administration area occupies 3.7 ha (9.1 acres), and is assumed to be a constant for all waste form and facility design and operation alternatives considered in this environmental impact statement. The area of the land committed for waste disposal--that is, the land actually containing disposed radioactive waste--varies according to the alternatives considered. The remaining acreage includes the operational area and the buffer zone as well as any excess land within the disposal area used for roads, working areas, and so forth. The acreage purchased by the site operator is assumed to be turned over to state ownership which is then leased back to the licensee.

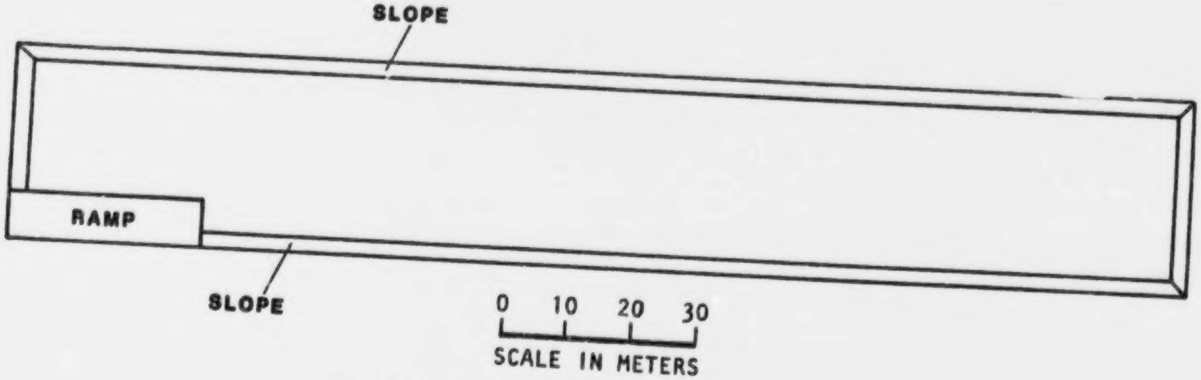
The site is surrounded by a 2.4 m (8 ft) high chain-link fence topped with three strands of barbed wire. A 2.4 m high fence also separates the administration area from the restricted area. Access to the disposal site is via a state highway running close to the site from which two short gravel roads lead onto the disposal facility. There are no rail facilities at the site. Incoming waste delivery and employee vehicles enter the site through one of two gates located in the administration area. These gates are locked at night and at other times such as holidays when the site is not being operated. Access to the restricted area is controlled by security check points near the gates in the fence separating the administration area and the restricted area.

For security purposes, a narrow gravel road runs alongside the inside of the fence surrounding the restricted area. Other onsite gravel roads wide enough to accommodate two small vehicles lead to the active disposal areas and are constructed by the licensee as needed. A lighting system is provided in the operational and adminstration area. There are no other lights installed in the interior of the restricted area.

The disposal area at the reference facility includes a number of disposal trenches. The reference disposal trench is assumed in this appendix to have average surface dimensions of 180 m (591 ft) long by 30 m (100 ft) wide, and have an average depth of 8 m (25 ft) (Figure C.3). However the length and width of the disposal trenches may vary somewhat (about ± 10 m). The rather large trench sizes assumed in this appendix are representative of trends at existing disposal sites.


CAP
COVER
WASTE
WASTE
WASTE
0 5 10 15 20
SCALE IN METERS

TYPICAL TRENCH CROSS-SECTION


SLOPE
RAMP
SLOPE
0 10 20 30
SCALE IN METERS

PLAN VIEW OF TYPICAL TRENCH

Figure C.3. Typical Trench Details

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The site soils are cohesive, but not cohesive enough to allow vertical-walled excavations. Therefore, trench wall slopes of 1 horizontal to 4 vertical (1:4) are employed. In some circumstances, (e.g., when extensive sloughing occurs) more gentle slopes are employed. The trenches are separated by 3 m thick walls. These inter-trench walls are able to support only light vehicles. Other vehicles, including heavy construction and transport vehicles, require a more substantial substrate.

As a trench is constructed, the locations of the four corners of the trench are surveyed and referenced to a bench mark. An approximate one degree slope is provided in the bottom of a trench from end to end and from one side toward a 0.6 m x 0.6 m (2 ft x 2 ft) gravel-filled drain. The drain runs the entire length on the lower elevation side to provide for collection of any liquid drainage that might occur. A gravel-filled sump is located at the low corner of the trench. A ramp leading into the trench is constructed at the low end of the trench opposite the drain. This ramp is only wide enough for a single vehicle to use at one time. A layer of sand approximately 0.3 m (1 ft) thick covers the entire floor of the trench. This allows any water entering the trench to drain away from the emplaced waste packages, so that waste packages will not be sitting in standing water.

Each trench is equipped with a minimum of three 0.15 m (6 in) diameter polyvinyl chloride (PVC) standpipes located within the French drain and standing along the sidewalls of the trench. The bottom 3 feet of each standpipe is fitted with either a slotted PVC pipe screen or a wound mesh PVC screen. Two of the three standpipes are located at each end of the excavation. The third standpipe is usually located at the trench midpoint (also standing in the drain). These PVC standpipes function as observation wells or sumps.

C.2.2.2 Support Facilities and Structures

The support facilities include: (1) an administration building, (2) a health physics/security building, (3) a warehouse, (4) a garage, (5) a waste activities building, and (6) a storage shed. All structures at the site are one-story metallic structures on concrete pad foundations. The functions of each of the support facilities are described below.

Administration. The administration building contains office space for site management and other administrative personnel working at the site. The activities performed within this building include coordination of waste shipments to the site, billing customers, and other routine file work. Site records are also stored within this building.

Health Physics/Security. The health physics/security building serves as the focal point for the majority of disposal activities at the site. This building houses a security section, a counting room, health physics offices, a change room/locker room, a lunch area, and a supply room. The health physics and security personnel are housed in the same facility because many of the functions performed by these personnel are complementary. Security personnel check both site personnel and visitors into and out of the site through a centrally located checkpoint. The health physics personnel have the prime responsibility for checking vehicles into and out of the disposal area. All persons leaving the site must pass through a frisker station to check for contamination which may have been picked up onsite. A safety decontamination shower is located adjacent to the frisker location. Emergency equipment such as safety ladders, respiratory equipment, and anti-contamination suits are stored in the vicinity of the frisker station. The employee change/locker room down the hall from the health physics offices includes both a street clothes ("clean") and work clothes area. Showers are also located in this section of the building.

Warehouse. The warehouse is used to store supplies used onsite. This facility is located within the administration area so that delivery trucks need not enter the disposal area. Among the stored items in this warehouse are cables, hooks, drums, bags, and other miscellaneous hardware. Casks and site vehicles are stored in the operational area.

Garage. Only vehicles and equipment that have been surveyed and decontaminated to within specified limits are allowed to use the garage. The garage is large enough to hold two vehicles at a time for maintenance. Mechanic's tools, spare parts, oil, and fuel (adjacent to the building in underground tanks) are also stored in the garage.

Waste Activities. This building houses several functional araes, including: (1) a large item decontamination bay, (2) a control room for the decontamination bay, (3) a liquid treatment system, (4) a waste solidification, packaging, and overpacking area, (5) a supply room, and (6) a small waste storage area.

The decontamination bay is used for washing down (decontaminating) large pieces of equipment (including trucks if necessary) through the use of a high-pressure recirculating water supply system. Contamination levels in these decontamination liquids are generally quite low; however, water treatment is applied to recirculating fluids. Small-scale decontamination of tools and other small items may be accomplished within the solidification staging area. The solidification area includes batch concrete mixing equipment for solidification of small quantities of low-activity liquids. A small storage area is provided for occasional temporary storage of shipments received from common carriers. A loading dock is located along the southern corner of this building.

Storage Shed. A storage shed is used to store supplies and miscellaneous tools used at the disposal trenches. This shed is portable and is usually located close to the active disposal trenches.

C.2.2.3 Site Administration

The organizational structure of the reference disposal facility is described in this section. One of the primary functions of the site organization is to provide managerial controls for the safe handling of radioactive materials at the disposal facility. Table C-1 contains a list of typical site personnel. The following discussion is conducted assuming a large disposal facility accepting on the order of 50,000 m^3 of waste per year. A more streamlined organization would be expected at a smaller site.

Corporate Management. The disposal facility is assumed to be operated for profit by a small corporation which is engaged in other nuclear-related business activities in addition to operating the disposal facility. The home office of the corporation is located in another state. Overall control of radiation health and safety at the corporate level is under the control of the senior radiation safety officer, who is responsible for conducting periodic reviews of site operations for compliance with health and safety regualtions and license conditions, including periodic site inspections and audits.

Site Management. Operations at the disposal facility are under the overall direction of the site manager and the assistant site manager. Beneath this level of site management, the administration of the disposal facility is organized into five parallel divisions: site operations, health physics, quality assurance (QA), administration, and security.

The site operations division controls such activities as trench construction, waste handling and disposal operations, and site groundskeeping and maintenance, and is under the direction of the operations manager. The site foreman assists the operations manager and is in daily contact with the site labor force. The foreman is responsible for work assignments, crew coordination, maintaining proper operating readiness of equipment, and general supervision of onsite burial and maintenance operations. The work force, under the control of the operations manager and site foreman, is composed of heavy equipment operators, skilled laborers, and unskilled laborers. Heavy equipment operators are responsible for the operation and routine maintenance checks of equipment used at the site for waste disposal and maintenance operations. Skilled laborers have a variety of functions at the site, including maintenance of site buildings and site property, operation of agriculture equipment, some heavy equipment operation, and some handling of waste material. Some of these laborers double as equipment mechanics when necessary. Unskilled laborers perform manual waste handling activities and other general support functions including maintenance of the facility buildings and grounds.

The health physics division is under the direction of the site radiation safety officer (RSO), who is responsible for ensuring that proper radiation work procedures are used, that adequate monitoring for radiation hazards is provided, and that personnel training, equipment, and techniques provide control of radiation exposure during site operations. Besides the radiation

Table C-1. Typical Disposal Facility Personnel

Senior Staff

Site Manager
Executive Secretary
Site Radiation Safety Officer
Assistant Site Manager
Foreman
Operations Manager
QA & Safety Supervisor
Site Engineer
Office Manager
Security Chief
Librarian (Records)
Customer Service Coordinator
Contracts Coordinator
Personnel Manager
Regulatory Affairs Manager

Support Staff

Junior Engineers
Waste Shipment Schedulers
Billing/Accounting Personnel
Security Personnel
Secretarial Personnel

Workers

QA Technicians
Radiation Safety Technicians
Heavy Equipment Operators
Skilled Laborers
Unskilled Laborers

safety controls, the RSO is also responsible for coordination of the site-safety training programs with the QA division, and for implementation of site emergency plans, procedures, and drills. The RSO reports directly to the site management as well as to corporate management, particularly the senior radiation safety officer. Routine health physics functions such as environmental and personnel monitoring are conducted by health physics technicians under the supervision of the RSO. Their duties also include inspections of incoming and outgoing vehicles as well as site surveys for control of radioactive contamination.

The primary responsibility of the security division is to control personnel and vehicle access to the site and to preclude potential theft of site tools or radioactive materials. The security force is under the control of a security chief and performs such functions as checking personnel and visitors into and out of the disposal site, conducting periodic patrols of the grounds and the site perimeter, and maintaining communications with law enforcement and other offsite emergency personnel. Like the site RSO, the security chief has direct lines of communication with corporate management, particularly the senior radiation safety officer.

The administration division is responsible for routine office work under the supervision of the office manager, including coordinating shipments, maintaining records, and billing customers. This division can be conceptually divided into three basic sections: records, customer service, and contracts.

Records are kept by a site librarian who maintains files and performs other functions including document reproduction, data recall, and coordination of routine reports. The secretarial staff functions as typists, file clerks, bookkeepers, and receptionists as needed by the various departments.

The customer service section coordinates the delivery of radioactive material to the site, schedules shipments, assesses charges for disposal services, and bills customers. The customer service section also informs customers of current disposal requirements and facility services which can be provided. Payment and accounting for routine site expenses are also handled by this section. The contracts section consists of a contract coordinator who works with corporate management and other site operational divisions to obtain needed outside services such as laboratory analyses, heavy equipment rental, transportation services, and utilities. The contract coordinator also arranges the use, as necessary, of outside consultants such as a registered surveyor.

The quality assurance (QA) program at the site is run by a QA supervisor who has three technicians under his supervision. The function of this division is to maintain compliance with applicable regulations, license conditions, and approved operational procedures. The QA supervisor has stop-work authority over site operations. Some of the site operations which are monitored by QA technicians include: trench construction, closure, site maintainance, waste disposal, equipment maintenance, and legal and procedural compliance by waste shippers and site personnel. Safety inspections, reviews of maintenance records, and training of site personnel are also included in their duties.

C.2.2.4 Site Operations

Site operations discussed in this section include: waste receipt, inspection, handling, storage, and disposal; radiation and contamination control; site groundskeeping and maintenance; environmental monitoring; security; recordkeeping and reporting; and quality assurance.

Waste Receipt and Inspection. Shipments of radioactive waste arrive by truck (generally as sole use shipments but occasionally via common carriers) and are processed onto the site on a first come, first served basis. Accompanying the shipments are manifest documents--termed radioactive shipment records (RSRs)--which describe the content of the shipment. Arriving shipments are inspected for compliance with applicable federal regulations and waste acceptance criteria established as conditions in the disposal facility license. Applicable federal regulations include those promulgated by NRC in 10 CFR Parts 20, 61, and 71, as well as those promulgated by the Department of Transportation (DOT) in 49 CFR 170-179. These regulations include, for example, waste packaging requirements, labelling requirements, vehicle placarding requirements, and allowable direct radiation and removable contamination levels at accessible surfaces of transport vehicles.

Waste acceptance criteria at existing disposal facilities vary somewhat from site-to-site. For purposes of this appendix, those waste acceptance criteria which are assumed for the reference disposal facility include packaging criteria for liquid scintillation vials, absorbed liquids, and animal carcasses, as well as a limit on the amount of free-standing liquid allowed in waste packages. Other reference criteria included limits on the quantities of radioactivity that may be received and possessed onsite at one time prior to disposal as well as package and shipment quantity limits for special nuclear material. (Special nuclear material includes uranium-233, uranium enriched in uranium-235, and plutonium.)

The results of these inspections are recorded on radiation survey forms and summarized on the RSRs accompanying the waste shipments. Shipments found to be in compliance with federal regulations and license conditions proceed into the disposal area for unloading. Violations of transportation regulations are reported to federal and state authorities in compliance with federal and state regulations and license conditions. Damaged or leaking waste packages are identified and appropriate protective or remedial action is taken. Depending upon license conditions, damaged or leaking waste containers may be overpacked or repackaged, and either accepted for disposal or returned to the sender. If detected, free-standing liquids are removed and solidified. Activities such as overpacking and solidification are performed at the waste activities facility.

Waste Storage. Generally, waste received at the site is disposed within a few days. Waste that must be temporarily stored is generally left in transport vehicles. There may be a reason, however, to temporarily store a few packages in a designated storage area, as when waste packages arriving by common carriers are stored temporarily. Since it often takes considerable time to process a waste transport vehicle into and out of a disposal site, it is sometimes more convenient to drop off waste packages (and accompanying paperwork) received from common carriers at the site storage area. The waste can then be disposed at a later time.

An added storage requirement exists for wastes containing special nuclear material. License conditions require that any single shipment of special nuclear material must be stored at least 12 feet from any other package containing special nuclear material.

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Waste Disposal. Waste is emplaced in the trench and backfilled with soil removed during trench excavation. Typically, waste packages are emplaced with the aid of construction equipment such as cranes and forklifts, using a combination of stacked and random disposal. Waste packages such as wooden boxes or steel bins having rectangular dimensions are generally stacked in place while low activity drummed waste is generally emplaced in a more random manner. Special care is taken during emplacement of higher activity waste such as high activity ion exchange liners to ensure operational safety. This combination of random and stacked disposal is termed "random disposal" in this report (to distinguish it from a placement alternative of fully stacked disposal) and results in a trench volume efficiency of about 50%. License conditions prohibit uncovered waste from extending more than 100 feet beyond the backfilled portion of the trench. License conditions also require that backfill operations commence immediately if radiation readings greater than 100 mR/hr are recorded at the trench boundary, and continue until radiation levels are reduced below 100 mR/hr. License conditions prohibit waste packages from standing in or being placed in water, so waste disposal commences at the high end of the trench and works down toward the lower end. Rainwater falling within the open trench and contacting the uncovered waste packages drains away to the lower end of the trench. Rainwater collecting in the lower end of the trench is then removed as necessary and treated by such methods as solar evaporation or solidification.

Waste is emplaced to within approximately 1 m of the top of the trench. Earthen fill is then backfilled into the trench. A 1 m thick cap composed of soil is then placed upon the backfill and is mounded. No special compaction is performed on the backfill and cap other than that provided by trucks and heavy earth-moving equipment driven over the top of the cap. The cap is then covered with natural overburden material as necessary to provide good drainage characteristics and according to the final contours planned for the site surface. The overburden is then reseeded to promote growth of a short-rooted grass cover.

Similar to the storage requirements discussed above, an additional requirement exists for disposal of wastes containing special nuclear material. License conditions require that each package of waste containing special nuclear material be disposed in such a manner as to have a minimum of 8 inches of earth (or wastes not containing special nuclear material) in all directions from any other package containing special nuclear material.

Following trench capping, the disposal trenches are each marked with a monument which is inscribed with the following information:

- A trench identification number;
- Total trench activity of radioactive material in curies, excluding source and special nuclear material; mass of source material in kilograms: and mass of special nuclear material in grams;
- Date of completion of waste disposal into the trench; and
- Volume of waste in the trench.

In addition, each of the four top corners of the disposal trench is marked with a marker stone.

During waste handling and disposal, operations are monitored to ensure radiation safety. After the transport vehicle is unloaded, it is again surveyed for contamination and decontaminated, as necessary, prior to leaving the restricted area. The results of the survey are recorded.

Site Groundskeeping and Maintenance. Groundskeeping includes both the upkeep of grounds and the maintenance of external building surfaces. The purpose of groundskeeping is to promote site integrity by maintaining proper contour and soil conservation practices, by properly maintaining external structures and site systems, and by overseeing closed burial trenches in an efficient manner. Groundskeeping activities include contouring of the ground surface, emplacement of a soil cover material such as grass, fertilizing, mowing, etc.

A site maintenance program entails routine inspection of site surfaces and fences for trench settlement, gullying, damage, debris, etc. Repairs are made as necessary.

An important part of the reference facility site groundskeeping and maintenance program is surface water management. A surface water management program is site-specific (i.e., is dependent on each site's topography, amount of rainfall, etc.), but its overall purpose is to divert surface water resulting from precipitation away from open trenches and to allow the surface water to flow offsite in a manner which will minimize erosion. The reference disposal facility is assumed to utilize low berms around open trenches to help accomplish this.

C.2.2.5 Site Safety, Radiation, and Contamination Control

The site operator will have a number of programs, operations, and procedures to ensure site safety and to minimize potential offsite releases of contaminants. Many of the site procedures are utilized for a combination of reasons. For example, strict monitoring for compliance with DOT transportation regulations serves to help reduce site personnel exposures and site contamination levels in addition to reducing potential transportation impacts. Controlling site contamination to low levels reduces personnel exposures as well as minimizes potential offsite releases through liquid or airborne pathways.

The remainder of this section describes methods used by licensees to maintain site safety and to control radioactive materials at the disposal facility. This section has been divided into five subsections as follows:

- personnel radiation monitoring;
- site radiation and contamination control;
- industrial safety;
- abnormal or emergency situations; and
- training.

Personnel Radiation Monitoring. Personnel radiation monitoring at the reference facility includes use of personnel monitoring devices, periodic internal monitoring, and administrative controls to ensure radiological safety.

Monitoring devices are worn by all site personnel who may become occupationally exposed to ionizing radiation. A long-term record of cumulative personnel exposures is maintained through the use of film or thermoluminescent dosimeter

(TLD) badges. These are replaced, analyzed, and the resulting exposures reviewed and recorded on a periodic basis (usually on a monthly or quarterly schedule). In addition, monitoring badges are replaced and analyzed whenever there is reason to believe that an employee may have received an unusually high radiation dose. Pocket dosimeters are also worn by site personnel and are used to provide an indication of radiation exposures over shorter time periods. These basic monitoring devices may, depending upon the circumstances, be supplemented by additional equipment such as electronic dose rate meters, finger or wrist monitoring badges, and/or continuous air samplers.

A periodic internal monitoring program is maintained for exposed individuals as a supplement to use of personnel monitoring devices (film badges, TLDs, and dosimeters). The internal monitoring program consists of an annual gamma scan for the lungs and thyroid in addition to a semiannual comprehensive bioassay. An immediate gamma scan and bioassay collection are performed if there is reason to believe that a site worker may have inhaled or ingested contaminated material. The gamma scan and bioassay are normally carried out at a nearby diagnostic laboratory. If, through a site accident, a worker receives an open wound and the wound is suspected of having become contaminated, a radiation survey is also performed. The survey is performed for beta and gamma contamination, and also for alpha contamination if alpha emitting isotopes are suspected. Backup blood or wound swab samples are also collected and analyzed.

Administrative controls are used to maintain exposures to levels as low as reasonably achievable (ALARA). A baseline is established for each new site employee which includes that employee's previous radiation exposure history. This baseline, quantified by an initial gamma scan and bioassay, is used to establish the employee's body burden prior to working at the site, and allows an evaluation of additional exposures received at the site. Records of subsequent occupational exposures are maintained, and an employee's functions are typically rotated to preclude individual employees from receiving a disproportionate share of exposure.

Site Radiation and Contamination Control. The licensee conducts routine radiological surveys to detect removable contamination and fixed radioactivity, and to minimize the potential for spread of contamination or unnecessary exposure to radiation.

As discussed above, waste transport vehicles and waste packages arriving at the disposal facility are routinely inspected by health physics technicians for compilance with federal regulations and license conditions.

Shipments found to be in compliance with transportation regulations and license conditions proceed into the disposal area for unloading. Violations of transportation regulations are reported to federal and state authorities. Damaged or leaking waste packages are identified and appropriate protective or remedial action is taken. Vehicle offloading operations are monitored by health physics technicians equipped with portable radiation survey equipment. Radiation levels at the edge of the active trench are controlled to levels less than 100 mR/hr. Should radiation levels exceed 100 mR/hr, the trench is immediately backfilled with earth or low-activity waste until radiation levels drop below 100 mR/hr. Waste transport vehicles leaving the disposal area are again inspected, surveyed, and decontaminated as necessary to comply with transportation regulations.

Routine housekeeping activities carried out at the site to minimize and control the spread of contamination include periodic surveys of site grounds, buildings, and equipment. This survey program is supplemented by the environmental monitoring program discussed below. Surveys or contamination control are also carried out whenever a site area or piece of equipment is used in a controlled area known to be contaminated or suspected of contamination through such possible events as a small spill.

Personnel procedures to control contamination and minimize exposures also include the use of anticontamination clothing and personnel surveys. At the reference disposal facility, waste handlers and other site personnel who may come into contact with radioactive materials are required to wear anticontamination coveralls, gloves, and other items as necessary. These are replaced when contaminated, and the used items are disposed as radioactive waste or sent offsite to a radioactive laundry service for decontamination. Site protective clothing may be supplemented by additional equipment such as additional anticontamination clothing, controlled air suits, or respirators if the situation indicates their use. All persons are required to conduct personal surveys ("frisking") for contamination upon leaving the restricted area as well as any other time the person has reason to suspect that he has become contaminated. Safety decontamination showers are provided for use, and personnel decontamination, if needed, is supervised by the site radiation safety officer.

Industrial Safety. Radioactive waste disposal facility operators in the past have generally concentrated on radiation safety to the exclusion of a separate comprehensive industrial safety program. The radiation safety officer at the reference disposal facility doubles as the safety officer, and has one radiation safety technician assigned specifically to inspection of equipment and job safety practices and hazards safety control. A program of industrial safety paralleling the radiation program does not, however, exist at the reference facility. Rather, aspects of personnel safety such as use of protective clothing are generally meant to meet both radiation and industrial safety standards. For example, all radiological workers or personnel working with overhead equipment wear anticontamination coveralls, anticontamination gloves, protective hard hats, and safety-toe shoes. In addition, workers are trained to follow specific work rules and respond to standard signals and alarms used on site equipment and by supervisory personnel.

Abnormal or Emergency Situations. Procedures and specific actions are established at the reference facility to aid in quickly and safely handling abnormal occurrences or site emergencies. Abnormal occurrences may include events such as a minor injury or a minor spill of radioactive material. A minor injury may be addressed through use of first aid equipment contained in site vehicles and in the health physics offices. In the event of a minor spill, the radioactive material is recovered and disposed, and the area in which the spill took place is decontaminated.

Site emergencies are expected to occur much more infrequently than abnormal occurrences, but are also expected to be potentially more serious. Specific actions contained in site procedures for emergency situations may include efforts to minimize exposures and control the potential spread of contamination, use of additional anticontamination or respiratory equipment, communication with emergency or law enforcement agencies, notification of regulatory authorities, and filing follow-up reports.

Overall control of an emergency situation is the responsibility of the site RSO, who directs actions of radiation safety technicians and other site workers in addition to coordinating with site security. Site security personnel maintain communications with site personnel responding to an emergency as well as with offsite emergency organizations. A central communications control point is maintained in the site security office and from this point, site personnel can be directed in their actions via loudspeakers located in the administration and operational areas and by radio communications with site security vehicles. These are radio-equipped four-wheel drive vehicles containing emergency equipment such as respirators and anticontamination clothing. The radios can also be used to communicate with local offsite emergency services such as police, fire, and ambulance. A call tree is established at the facility for telephone communication with corporate and site management, site personnel, local offsite emergency services (police, fire, ambulance), federal emergency services (e.g., DOE regional coordinating offices for radiological assistance), and federal and state regulatory agencies.

In addition, existing federal regulations require timely notification of federal authorities for a number of types of abnormal occurrences or emergency situations, as well as follow-up reports. Specific notification and reporting requirements depend upon the severity of the incident.

At the reference facility, a checklist is used to ensure that initial notification, follow-up reporting, and follow-up action implementation are completed within time limitations specified in the regulations. Data from these incident reports are used to evaluate and improve the site quality assurance and industrial safety programs.

The historical record does not contain evidence of accidents resulting in acute releases of radionuclides which would present a hazard to the public health and safety. Nonetheless, specific site procedures addressing a number of different types of radiological and nonradiological emergencies are developed at the reference facility and drilled at least annually. These site procedures include planning for such potential events as major spills; treatment of irradiated, contaminated, injured, or contaminated and injured personnel; fires; bomb threats; and civil disorders.

Incidents such as a major spill would generally be expected to involve a transport vehicle or possibly a waste container accidently dropped from a crane. In this case, steps are taken to rope off and contain the contamination to a small area. The vehicle can then be decontaminated in a manner which minimizes personnel exposure.

In general, treatment for trauma, shock and hemorrhage takes precedence over personnel decontamination procedures and treatment of possible symptoms from irradiation. If an injured worker is in a high radiation environment, he is immediately removed from this environment concurrently with the administering of any other immediate life-saving actions that may be needed. The site is equipped with safety decontamination showers and first aid equipment, and some of the radiation safety technicians have additional training in emergency medical care.

Prior arrangements have been made with a local hospital to receive injured personnel. If necessary, a site health physics technician may accompany

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ambulances to assist hospital personnel in further decontamination of injured site personnel.

For potential onsite fires, the main concern is the possibility of generating airborne radioactive material and the spread of contamination. A number of hand-held fire extinguishers are located on site vehicles and at a number of (stationary) site locations. Generally, a fire in a trench could be quickly extinguished by backfilling with soil. Personnel involved in the emergency use respiratory equipment and anticontamination clothing as appropriate. Local offsite fire fighting agencies may be called upon to assist. Arrangements are made by the site RSO to periodically review elements of basic radiation safety with these agencies.

Training. Pursuant to 10 CFR Part 19 of the Commission's regulations, each disposal site employee receives instruction in the hazards and controls of radioactive materials commensurate with the worker's duties and responsibilities for handling materials, and with the extent of anticipated worker exposure. This is combined with training in operational procedures to provide a solid basis for safe onsite work. Supplementary training to deal with site emergency conditions is also carried out. The worker is instructed at initial orientation, in the classroom, and under actual working conditions in a variety of subject areas. Each employee is certified upon the successful completion of each training subject area, and training records documenting the type of training received and the resulting scores are kept for a 2-year period at the site. Periodic refresher training and recertification is required after initial qualification.

C.2.2.6 Environmental Monitoring Program

An environmental monitoring program is carried out at the reference disposal facility to monitor and control potential releases of radioactivity during site operations, to detect movement of radionuclides from the disposal trenches to the environment, and to provide information as to the long-term containment of radioactive waste disposed at the site. This is accomplished through an environmental sampling program in which samples are collected on the disposal site in addition to samples collected at a number of offsite locations. Potential exposure pathways to and possible impacts on individuals or local populations are also evaluated as part of the program.

Monitoring for potential long-term environmental releases is principally accomplished through groundwater samples obtained from both onsite and offsite wells, and from monitoring for the presence of water, if any, in trench sumps.

Monitoring for potential short-term operational releases of radioactivity is principally accomplished through collection of soil and vegetation samples, air samples, and samples collected to monitor direct gamma radiation levels. This portion of the environmental monitoring program is performed in conjunction and coordinated with the site survey program discussed above. Information obtained from the environmental monitoring program is used to improve the effectiveness of the survey program and vice versa.

Potential short-term releases are expected to be minimal since site operations mainly involve handling of packaged wastes. However, some minor airborne releases may occur from processing (e.g., solar evaporation) of precipitation

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collected in operational trenches. Some minor airborne releases may also occur through operations at the waste activities facility (e.g., vehicle washdown, waste repackaging, liquid solidification). These operations, however, are monitored by airborne sampling equipment. Experience has shown that operational releases can be minimized through strict contamination control practices.

A summary of a typical operational monitoring program is included as Table C-2. This program includes collection of groundwater samples, soil and vegetation samples, and air samples, as well as monitoring for direct gamma radiation levels. Results from individual measurements recorded in the environmental monitoring program are retained for the life of the site along with information on sampling location and date, sample size (e.g., wet/dry weight), sampling and analytical procedures, units of data, and precision and accuracy associated with individual measurements.

As implied by Table C-2, environmental monitoring programs at past and existing sites have mainly emphasized radiological analyses. Some nonradiological monitoring is also being performed, however, and this is expected to be more extensive in the future.

C.2.2.7 Security

The site security program is implemented not so much for radiation health and safety considerations, but particularly to protect the many thousands of dollars worth of equipment, buildings, and facilities located onsite. The security program at the case facility is assumed to include the following:

- Full-time security personnel and a security training program.
- Controlled access and exit from site areas including fencing and lighting, material gate passes, badge control, personnel and vehicle search procedures, lock and key control, etc.
- Radio and telephone communication ability with emergency and law enforcement agencies.
- Identification badges and dosimetry for site employees and visitors.
- Procedures for notifying site personnel and local authorities in the event of an emergency.

Security personnel are assumed to interact closely with health physics personnel and carry out a number of functions which include checking people and vehicles into and out of the site, periodic patrols of site areas and fences, helping to respond to emergencies, and maintaining communications. Access to the restricted area is limited to employees, authorized unescorted visitors and contractors, visitors with escorts, and federal and state inspectors who require access to perform their duties. Site security equipment includes a few four-wheel drive vehicles which are equipped with radios to communicate with each other and with a central security control point, and which also contain emergency equipment such as anticontamination clothing.

Table C-2. Typical Operational Environmental Monitoring Program

Sample Description	No. of Locations	Type	Media	Frequency of Analysis	Type of Analysis
External Gamma	50	Continuous	TLD	Quarterly	Exposure
Atmosphere	3	Continuous	Particulate Filter	Daily	Gross Beta-Gamma
			Particulate Filter	Weekly	Gamma Isotopic
			Charcoal Cartridge	Weekly	I-131
Soil & Vegetation	10	Grab	-	Quarterly	Gross Beta-Gamma Gross Alpha Tritium
Offsite Wells	5	Grab	H_2O	Semiannual	Gamma Isotopic Gross Alpha Tritium
Site Boundary Wells	10	Grab	H_2O	Semiannual	Gamma Isotopic Gross Alpha Tritium
Disposal Area Wells	15	Grab	H_2O	Quarterly	Gamma Isotopic Gross Alpha Tritium
Filled Disposal Trench Sumps	58	Grab	H_2O	Monthly	Gamma Isotopic Gross Alpha Tritium

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*Trench sumps are checked on a monthly basis. Analysis would take place only if water was determined to be present in a sump.

C.2.2.8 Recordkeeping and Reporting

A number of records are maintained at the site to cover the areas required by regulation, operational control, and for future use. Records which are assumed to be maintained at the facility include:

- Personnel exposures
- Waste receipt and disposal records
- Personnel training records
- Records for the QA program
- Environmental monitoring data
- Operating procedures
- Records of site surveillance and monitoring

In general, administrative records such as personnel files, internal office memos, preliminary designs and budgets, and other records of this type are not permanent and are assumed not to be retained longer than 3 years.

A number of reports are made to regulatory agencies. Reporting requirements include periodic site inventory data, notices of shipper noncompliance, notices of abnormal events or license violations, personnel exposures, and environmental monitoring results. The licensee also complies with the reporting requirements of 10 CFR 70.54 (Nuclear Material Transfer Reports) for special nuclear material.

C.2.2.9 Quality Assurance

The quality assurance (QA) program at the site functions as a parallel department which provides quality control and training support to disposal operations. The QA personnel are not only familiar with the operating procedures, maintenance requirements, safety rules and basic radiation work procedures, but also have the responsibility to recommend improvements and coordinate the site training program. QA documentation is streamlined but is intended to provide adequate information to identify and correct substandard items. The QA program includes the following areas:

- personnel monitoring
- emergency drills and equipment
- contamination control
- working procedures
- site maintenance
- site groundskeeping
- waste receipt, inspection, storage, and disposal
- radiation instrument care and calibration
- environmental monitoring
- security
- construction of disposal trenches
- closure and stabilization
- recordkeeping

In addition, a management audit program is carried out at least quarterly to maintain high standards of radiological control and safety and to ensure compliance with federal, state, local, and site license requirements. The program includes a review of operating procedures and past exposure records, facility inspections, and surveillance of work being performed. The senior radiation safety officer is directly responsible for the implementation of the audit program.

C.2.3 Postoperational Activities

This section briefly describes the assumed actions taken by the licensee to close the reference disposal facility and to prepare the facility for long-term care by the site owner. Long-term care activities by the site owner are also briefly described.

C.2.3.1 Reference Disposal Facility Closure

Final closure of the reference disposal facility is assumed to require approximately 1 year and mainly involves dismantling and decontaminating site buildings, disposal of wastes produced during dismantlement and decontamination operations, and final site seeding and contouring. This is followed by a somewhat indefinite period of time prior to license termination during which the grass cover over the final trenches is established, the site is inspected prior to transfer to the site owner, etc.

Three of the six buildings on the reference disposal facility--the administration building, the health physics/security building, and the site warehouse--are located in the administrative area of the site and should be free of contamination. The administration building and the warehouse are dismantled and sold for salvage. The health physics/security building is left standing for use by the site owner during the institutional control period. Of the remaining three site buildings, only the waste activities building is expected to have appreciable levels of contamination. This building is decontaminated to the extent practical and is demolished, as is the site garage. The site shed is decontaminated as necessary and is left standing onsite for further potential use by the licensee and the site owner. Waste produced during dismantlement and decontamination operations is disposed at the disposal site.

For the reference facility, there is assumed to be no effort to recontour the disposal site land. The trench covers are left mounted. The final disposal trenches are filled, capped, graded, and seeded with a grass cover.

During this time period, the licensee makes a final survey of the disposal area to determine direct radiation levels. All parts of the disposal area are certified as having radiation levels at essentially background levels. A few hotter spots may be observed but these are filled with overburden as necessary to reduce the radiation levels to background.

C.2.3.2 Site Surveillance

This time period is of variable length, and occurs following site closure activities. During this period, the licensee would still be responsible for all site maintenance and environmental monitoring activities. This responsibility would be maintained by the licensee until the license is transferred to the site owner.

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C.2.3.3 Institutional Controls

At this point, the disposal license with the site operator is transferred to the site owner. For this appendix, the site owner is assumed to be a state agency. Activities which take place during the institutional control period are carried out under control of the site owner and include site inspection, maintenance, and monitoring. Maintenance activities may be required during this time period, mainly involving repair of slumping, subsidence and other disposal trench instability problems. The required amount of maintenance will vary depending upon the waste volume, waste form, and disposal practice. During this phase, environmental monitoring of the disposal facility continues.

C.3 DISPOSAL ALTERNATIVES

This chapter is organized into sections addressing alternative disposal cell designs, disposal cell cover designs (caps), stabilization techniques, backfill materials, operational procedures, and post-operational considerations.

C.3.1 Alternative Disposal Cells

This section describes the alternative disposal cell concepts that are considered in this report.

C.3.1.1 Trench Disposal

The reference disposal facility discussed in Chapter C.2 is assumed to be located in a humid environment and is furthermore assumed to utilize disposal trenches having relatively large dimensions -- i.e., 30 m wide by 180 m long by 8 m deep. These assumptions are believed to be reasonable. Most low-level waste has been, and will continue to be generated in humid climatic areas, and regional or compact disposal of waste implies that such waste will be disposed in humid climactic areas. In addition, the assumed large trench sizes reflect the trend that has taken place over the last several years at existing commercial disposal facilities. Finally, a large disposal facility implies that it will accept waste from a large, multistate region, which also implies that a large area will be available for site selection. Thus, it is plausible that sites can be located having relatively long distances to ground water tables (e.g., on a topographical high point).

Given the current trends in the compacting process, however, it is apparent that there may be sites that only accept waste from one or perhaps a few states. There may be a number of sites located in arid or semi-arid environments. There may also be disposal facilities located at sites having moderately high water tables. Given the above, three variations in trench design are considered in this report. These variations include:

A. Small trenches, humid environment (10 m x 60 m x 6 m.);
B. Large, deep trenches, arid environment (30 m x 180 m x 14 m); and
C. Small trenches, arid environment (10 m x 60 m x 8 m).

For small disposal facilities, disposal trenches having small dimensions may be more convenient to construct and employ. One reason is that using the small trenches reduces the length of time that the trenches will be open to the elements. For Variation A, then, the disposal trenches are assumed to

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have nominal surface dimensions of 10 m by 60 m. The walls are again assumed to slope at a 1:4 ratio, resulting in as-excavated trench floor dimensions of 7 m by 57 m. The assumed 6 m depth reflects the possibility of moderately high water tables at the site location. Otherwise, the disposal facility trenches are generally assumed to be designed in the same manner as those in the reference disposal facility. A gravel drain is located along one side of the trench and a layer of sand covers the bottom of the trench. Unlike the reference trench, however, only two stand pipes per trench are assumed. The waste is emplaced (random disposal) up to 1 m below the top of the trench, which is then backfilled with indifferently compacted soil. A 1 m (minimum) soil cap is then emplaced which is shaped for drainage and planted with short rooted vegetation. Four corner markers are emplaced, as is a monument describing the trench contents. The spacing between the individual disposal trenches is assumed to be 3 m.

Both Variations B and C are designed assuming that the disposal trenches are located in arid environments where there will be no possibility of problems with accumulation of water within disposal trenches. The major difference between the two variations is the assumed trench size.

Trench Variation B is believed to be most appropriate for large disposal facilities and for sites at which the soils are cohesive. In this variation, the nominal surface dimensions of the trenches are again assumed to be 30 m by 180 m. However, the walls are assumed to slope at a 1:4 ratio to a trench depth of 14 m, resulting in nominal floor dimensions of 23 m by 173 m. The greater trench depth (14 m) is typical of existing practice at both the Beatty and Richland commercial disposal sites. Given the trench location in an arid environment, the gravel drain, sand layer on trench bottoms, and standpipes are assumed to be deleted. The ramp leading into the trench is also deleted, since it is believed that the trench will be generally too deep to allow convenient and safe waste emplacement from inside the trench. Waste emplacement will be performed using slings and cranes operated from outside the trench.

In operation, the waste is again assumed to be disposed randomly (disposal efficiency = 0.50) to a height of 1 m below the top of the trench. The trenches are then assumed to be backfilled with native soil and capped with a 1 m (minimum) layer of natural soil. Compaction is performed in an indifferent manner. Given the arid nature of the site, the trench caps are assumed to be stabilized using a 0.15 m (6 inch) layer of rocks and cobbles. This is the practice at the Richland commercial waste disposal facility and is believed to be a logical alternative to short-rooted vegetation, particularly given the difficulty expected in maintaining the vegetation without regular watering.

After capping, four corner markers are again assumed to be emplaced, as is a monument describing the trench contents. The spacing between the trenches is assumed to be 3 m.

Trench Variation C might be used for a small site, where the annual volume of received waste may not justify the more efficient land use afforded by larger trenches. For example, only a limited amount of heavy construction equipment may be available. There may also be a desire to limit the time that a trench would be open. In any case, the surface dimensions of the trench are assumed to be 10 m x 60 m, and, at a 1:4 wall slope and an 8 m depth, this results in nominal trench floor dimensions of 6 m by 56 m. Like Trench Variation A, a

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ramp is built at one end of the trench along one side of the trench wall. Like Trench Variation B, there is assumed to be no gravel drain, sand layer, or standpipes.

Operation of the disposal facility is assumed to be carried out similarly to that for Trench Variation B. The waste is emplaced randomly (disposal efficiency = 0.5) and then backfilled and capped with natural soil. The cap is stabilized with a 6-inch layer of rocks and cobbles, and markers are placed at the filled trench corners. A monument describes the trench contents.

Significant parameters which can be used to characterize the above disposal variations include the available disposal volume per trench, the actual waste disposal volume, the waste emplacement efficiency (EMP), the volumetric disposal efficiency (EFF), the surface utilization efficiency (SEF), the disposal cell depth (DPT), and the disposal cell thickness (DTK). These factors are defined in Table C-3.

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Table C-3. Definitions of Some Disposal Parameters

Parameter	Definition
Available disposal volume	Available disposal space (m^3), not considering backfill; however, factors such as space taken up by a ramp, sand base, maximum height of waste, etc., are considered.
Actual disposal volume	Actual volume of waste per disposal cell (m^3); equivalent to available disposal volume times waste emplacement efficiency.
Waste emplacement efficiency (EMP)	Volume of waste actually emplaced in a disposal cell divided by the available disposal space within the cell (dimensionless).
Volumetric disposal efficiency (EFF)	Volume of available disposal space in a disposal cell divided by disposal cell surface area (m). The surface area of the disposal cell is the horizontal plane that is in contact with the waste and is closest to the ground surface.
Surface utilization efficiency (SEF)	The total disposal area encompassing the disposal area, including spacing between cells, divided by the total surface area of the disposal cells (dimensionless). The surface area is assumed to be that defined above for EFF.
Disposal cell depth (DPT)	The depth of the top of the disposal cell below the top of the final disposal cell cover (m).
Disposal cell thickness (DTK)	The vertical distance (m) between the bottom and top planes of the disposed waste (the maximum height to which waste can be placed in the disposal cell).

Using the above assumptions, and also assuming a 3-meter spacing between disposal cells, a list of reference (conceptual) values for the above parameters is presented below. Also listed is the total surface area of the disposal area assuming disposal of 1,000,000 m^3 of waste. This latter is calculated as the waste volume (1,000,000 m^3) divided by the product EMP x EFF x SEF.

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Parameter	Trench Variation			
	Ref.	A	B	C
Avail. Disp. Vol. (m^3)	32,979	2,180	60,191	2,950
Act. Disp. Vol. (m^3)				
EMP = 0.5	16,489	1,090	30,096	1,475
EMP = 0.75	24,734	1,635	45,143	2,213
Vol. Disp. Eff. (m)	6.23	3.88	11.37	5.25
Surf. Util. Eff.	0.88	0.69	0.88	0.69
Disposal Cell Thickness (m)	6.7	4.7	13	7
Disposal Cell Depth (m)	2	2	2	2
Surf. Area (1,000,000 m^3 of waste)				
EMP = 0.5 (m^3)	366,170	750,890	200,630	554,830
(acres)	90.5	185.5	49.6	137.1
EMP = 0.75 (m^2)	244,110	500,600	133,750	369,890
(acres)	60.3	123.7	33.1	91.4

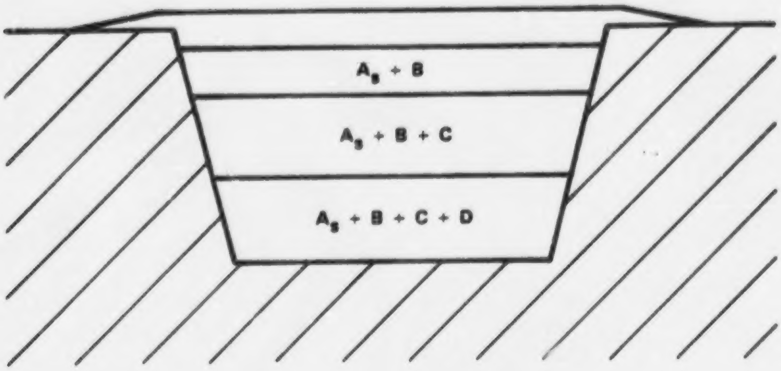
The reference values for the above parameters are used when the disposal cell is filled with only one class of waste. However, consider a case in which a disposal cell is filled with several waste classes as illustrated in Figure C.4.

In this case, there is assumed to be disposal depth requirements for various waste classes. Class A (stable) and Class B waste have no depth limitations, Class C must be disposed with at least 5 m cover of lower activity waste or other material such as soil. Class D waste must be disposed with at least 10 m of cover. The waste emplacement efficiency may be different for each class of waste within the disposal cell, as may be the disposal cell depth and the disposal cell thickness. This, plus other considerations, presents difficulties when calculating impacts.

To determine residual impacts from waste disposal, an approximation is made in which the entire volume of waste in a particular class is contained in a narrow band. This allows an "average" level of residual impacts to be calculated. An effective disposal cell depth and an effective disposal cell thickness is then calculated for each class as also illustrated in Figure C.4.

C.3.1.2 Trench Disposal with Additional Backfill

This alternative is wasteful of land, but illustrates one potential alternative for deeper disposal of low-level waste. It is most suitable for sites located in arid environments, since there is a strong possibility that the waste would otherwise intersect the water table. In this case, as in Trench Variation B in Section C.3.1.1, a large deep trench is first constructed in an arid environment. Waste is emplaced randomly in the trench up to a level 10 m below the ground surface. The remaining 10 m is backfilled with natural soil, capped with a minimum of 1 m of natural soil, and stabilized with a layer of rocks and cobbles.


A_s + B
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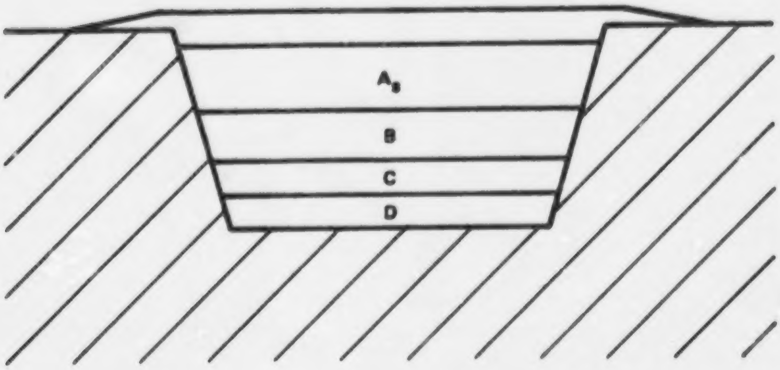
A_s + B + C + D

Actual



Conceptual

Figure C.4. Hypothetical Disposal Cell Containing Several Classes of Waste

Disposal of waste at greater depths helps to reduce potential impacts due to human, plant root, or animal intrusion into the waste. It has been estimated that a 10 m depth essentially completely removes the waste from root depths associated with food and livestock crops (Ref. 5). However, it can be seen that this alternative is wasteful of disposal space. The available disposal volume in a given trench is only 11,358 m^3, compared with 60,191 m^3 in Trench Variation B.

Assuming 1,000,000 m^3 of randomly disposed waste, about 176 disposal trenches would be required for this alternative, compared with only 33 disposal trenches for Trench Variation B.

A less costly alternative in terms of money and land use would be merely to layer the waste. That is, the higher activity waste could be preferentially placed at the bottom of the trench, with lower activity waste placed above the higher activity waste. This alternative is considered in Section C.3.5.3.

C.3.1.3 Trench Extension

As in Trench Variation B as discussed above, this alternative may be most suitable for disposal sites located in arid environments, where the possibility of intersecting the water table is minimal. The purpose of this alternative is to provide a way to achieve deeper disposal for a limited volume of waste. At a large disposal site, the probable use of large, deep disposal trenches will allow convenient emplacement of particular wastes at greater disposal depths. This may not be as convenient at a small site.

In this alternative, a standard 10 m x 60 m x 8 m disposal trench is first constructed as in Trench Variation C as discussed in Section C.3.1.1. Then a number of slit trenches are dug in the floors of some of the trenches. For this report, the slit trenches are assumed to be 6 m deep, which is a depth that can be easily reached with a backhoe. A 6 m depth also means that waste can be emplaced to the same depth below the surface (14 m) as in Trench Variation B above. The slit trenches are also assumed to be relatively short (only 10 m long) to allow disposal operations in the large trench to readily proceed. A 1:4 slope is assumed for the walls of the slit trenches, resulting in about a 1 m width at the floor of the slit trench and a 4 m width at the top of the slit trench.

In operation, waste will be emplaced carefully (stacked) within the slit trenches rather than being disposed randomly. The waste will furthermore be emplaced to within about 1 m of the top of the slit trench extension (the floor of the larger trench) and then backfilled up to the floor level of the larger trench. (This results in a disposal cell thickness of 5 m.) Waste will then be placed randomly in the larger trench and backfilled and capped as in Section C.3.1.1. (Note that the disposal efficiency of a slit trench extension is about 0.75 while the disposal efficiency of a larger trench is about 0.5.)

Assuming that the volume of waste to be disposed by this method totals only about 1% of the waste disposed at the site, this results in a total volume of 10,000 m^3, or 500 m^3/yr. At a total disposed waste volume of 71.6 m^3 of waste per slit trench extension, this means that only 7 slit trench extensions would be required per year, or 140 slit trench extensions over the 20-year life of the disposal facility.

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A more important question is the maximum amount of waste that can be disposed by this method, given the limitations of available floor area in the disposal trenches. The trench has nominal floor dimensions of about 6 m by 56 m, although the ramp cuts significantly into the available length. Realistically, given operational and handling limitations, only about three slit trench extensions can be excavated per large trench. These would be excavated lengthwise. Thus, assuming stacked disposal at 75% efficiency, only about 214.7 m^3 of waste can be disposed via slit trench extension per large trench. This implies that a limit in the amount of waste disposed by slit trench extension can be estimated as a fraction of the waste disposed via the large trench. This bounding fraction varies depending upon whether waste is emplaced stacked or randomly in the larger trench. For stacked disposal, the bounding fraction is 0.097, and for random disposal is 0.146.

It can be seen that this alternative is very flexible in that the level of planning required to dispose of wastes by this method is not very extensive. The slit trenches can be constructed more or less on demand, and in variable lengths and widths depending upon the dimensions of the particular waste packages to be disposed.

C.3.1.4 Auger Holes

Auger holes simply refer to a series of round holes drilled from the ground surface. There has been considerable experience with constructing, backfilling, and compacting auger holes at the Nevada Test Site (NTS) as part of testing and monitoring programs involving nuclear weapons. There has also been experience with auger hole disposal at Oak Ridge National Laboratory. Auger hole disposal is considered to be one of the prime alternatives in DOE's studies of potential greater confinement disposal (GCD) alternatives (Ref. 6). Auger holes are simple to construct, and auger holes up to about 8-10 ft in diameter can be dug using truck-mounted rigs. Otherwise, cranes are required at greater expense. An advantage of auger hole disposal is that it can reduce exposures to site workers during disposal operations, and will also help reduce impacts to a potential inadvertent intruder.

In this report, four different auger hole variations are considered, two of which are appropriate for a humid site while the other two are appropriate for an arid site. The variations include:

Auger Variation A:	1 m dia. x 6 m depth;
Auger Variation B:	1 m dia. x 30 m depth;
Auger Variation C:	3 m dia. x 6 m depth; and
Auger Variation D:	3 m dia. x 30 m depth.

Auger Variations A and C are appropriate for a humid environment, so a standpipe is assumed to be located along the side of each auger hole. Each standpipe is located in a 0.3 m gravel layer placed on the bottom of the auger hole, and terminates at the ground surface in standpipe access boxes. These standpipes are deleted for Auger Variations B and D, however, which are more appropriate for an arid site. The gravel layer is also deleted for Variations B and D.

In operation, waste is assumed to be stacked carefully within the auger holes, resulting in a disposal efficiency of about 0.75. Waste disposal is furthermore assumed to be carried out up to a level 1 m below the earth's surface, after which the auger holes are backfilled up to the top of the holes using native soil. Each group of auger holes is then capped with a minimum 1 m of native soil. The cap is then stabilized in Auger Hole Variations A and C by planting short rooted vegetation such as grass. For Auger Hole Variations B and D the cap is stabilized using a 6-inch layer of cobbles and rocks. A marker is placed above each auger hole, and each auger hole group is identified by a monument.

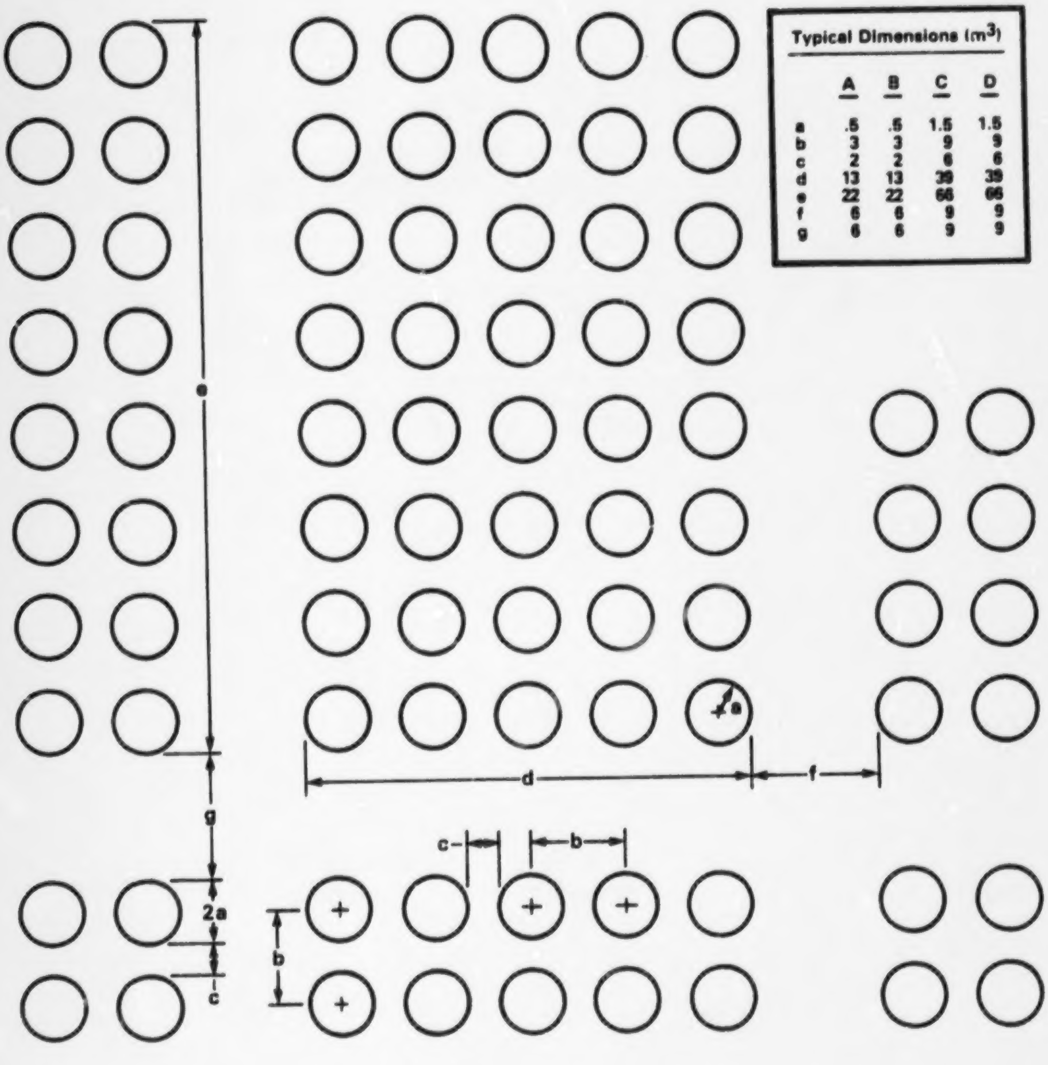
In this report, auger holes are somewhat arbitrarily assumed to be excavated in groups of 40 (Figure C.5). This approach allows for easier construction and disposal operations. In practice, groups would be constructed in a checkerboard arrangement as illustrated below, where each group is arbitrarily identified.

A	B	C
D	E	F

A typical sequence of construction could be A, E, C, D, F, B. This allows construction activities to be carried out separately from disposal activities.

Hole augering activities will tend to create unstable zones around each hole. Depending upon soil properties, these unstable zones may be large and characterized by fissures, cracks, and broken substrata. Thus, it is current practice to leave a minimum separation between two holes equivalent to three times the diameter of the augered hole. Thus, the dimensions depicted in Figure C.5 are assumed for each auger variation.

Based on the above assumptions, a list of appropriate disposal technology parameters is presented below. These are defined in Section C.3.1.1 and include the available disposal volume, the actual disposal volume, the volumetric disposal efficiency, the surface utilization efficiency, and the waste thickness. Also included is the total surface area of the disposal area assuming disposal of 1,000,000 m^3 of waste.



Typical Dimensions (m^3)

	A	B	C	D
a	.5	.5	1.5	1.5
b	3	3	9	9
c	2	2	6	6
d	13	13	39	39
e	22	22	66	66
f	6	6	9	9
g	6	6	9	9

Figure C.5. Auger Hole Layout and Dimensions

Parameter	Auger Hole Variation (40 cell group)			
	A	B	C	D
Avail. Disp. Vol. (m^3)	147.7	911.1	1329	8200
Act. Disp. Vol. (m^3) (EMP=0.75)	110.7	683.3	996.6	6150
Vol. Disp. Eff. (m)	4.7	29.0	4.7	29.0
Surf. Disp. Eff.	.059	.059	.079	.079
Disposal Cell Thickness (m)	4.7	29	4.7	29
Disposal Cell Depth (m)	2	2	2	2
Surf. Area (1,000,000 m^3 of waste)				
(m^2)	4.81E+6	7.79E+5	3.59E+6	5.82E+5
(acres)	1190	193	887	144

C.3.1.5 Auger Holes with Additional Backfill

This alternative is similar to the disposal alternative considered in Section C.3.1.2 (Trench Disposal with Additional Backfill). In this case, auger holes are assumed to be excavated as in Auger Hole Variations B and D discussed above in Section C.3.1.4. However, waste is only emplaced in the bottom 20 m of the auger holes. The top 10 m of each auger hole is backfilled with soil, and each auger hole group is capped with a 1 m (minimum) soil layer which is stabilized by a 6-in layer of cobbles and rocks. As above, each auger hole is marked with a marker, and each auger hole group is also identified using a monument.

As in Section C.3.1.2, this alternative is more costly than standard auger hole disposal, and also more wasteful of land. At a disposal efficiency of 0.75, each "deeper" auger hole can hold the following waste volume.

Auger Variation B with Additional Backfill: 11.8 m^3; and
Auger Variation D with Additional Backfill: 106 m^3.

As above, a more cost-effective practice would be to preferentially place the higher activity waste at the bottom of the auger shaft, with lower activity waste placed above the higher activity waste. This "layering" disposal concept is discussed in more detail in Section C.3.5.3.

C.3.1.6 Auger Hole Extension

This alternative is similar to the Trench Extension alternative considered above in Section C.3.1.3. It is assumed to be most suitable for an arid environment, due to potential problems with intersecting the water table at a site located in a humid environment, and offers a method of disposing of waste at greater depths at a site in which only small, shallow disposal trenches are utilized.

In this alternative, a number of shallow (6 m depth) auger holes are constructed at the bottom of some of the larger disposal trenches. In this report, it is assumed that the larger trenches correspond to Trench Variation C in Section C.3.1.1. These disposal trenches have surface dimensions of 10 m x 60 m and have 8 m depths. In operation, an auger rig will be driven into a trench, and one or more auger holes excavated. Waste will then be stacked within the auger holes (emplacement efficiency = 0.75) up to a 1 m level below the top of the auger hole. (The waste thickness is 5 m.) The auger hole will then be backfilled up to the surface of the trench floor, and then disposal operations will proceed in the trench as otherwise described under Trench Variation D in Section C.3.1.1.

For a site accepting 1,000,000 m^3 of waste over a 20-year operation, a reasonable assumption would be that no more than 1% of the waste would require deeper disposal using auger hole extensions. Assuming a 20-year operating life, this corresponds to 500 m^3 of such waste per year. Assuming auger hole dimensions corresponding to Variation A in Section C.3.1.4 (1 m dia. x 6 m depth), this results in 2.95 m^3 of disposed waste per auger hole, or 170 auger hole extensions per year. If auger hole dimensions corresponding to Variation C are assumed, however (3 m dia. x 6 m depth), 26.51 m^3 of waste may be disposed per auger hole. This would require only about 19 auger holes per year.

As in Section C.3.1.3, a more pertinent question is the maximum amount of waste that can be disposed using the auger hole extension technique, given the limitations in available floor area in the disposal trenches. Equally important considerations are the ease in which waste containers can be handled and emplaced. For 1 m diameter augers, it is conceivable that up to 28 augers can be handled with some measure of convenience. This corresponds to 14 rows of augered holes along each side of the trench, and results (assuming stacked disposal in the augered holes) in a maximum of about 82.5 m^3 of waste disposed via augered holes per large trench. For 3 m diameter augers, a reasonable maximum number of holes is believed to be about 5 per trench; this results in a maximum of about 132.5 m^3 of waste per trench.

As in Section C.3.1.3, a limit for waste disposal via auger trench can be estimated as a fraction of the waste disposed via the trench. This fraction varies depending upon the auger hole diameter and whether waste is stacked in the trench or disposed randomly. Calculated bounding fractions are listed below:

Variation	Bounding Fraction
1 m dia., stacked	0.037
1 m dia., random	0.056
3 m dia., stacked	0.060
3 m dia., random	0.090

Similar to the slit trench extension (Section C.3.1.3), this alternative is very flexible in that the level of planning required to dispose of particular

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types of wastes by this method is not very extensive. The auger holes can be constructed more or less on demand, and in variable diameters corresponding to the dimensions of the particular waste packages to be disposed.

C.3.1.7 Slit Trench

A slit trench is another disposal alternative which is useful for disposal of wastes having high surface radiation levels. It will also tend to somewhat reduce impacts to a potential inadvertent intruder. Slit trenches have been used for disposal of waste at both the Oak Ridge National Laboratory and the commercial disposal facility located near Barnwell, South Carolina.

In this report, two slit trench variations are assumed, depending upon whether the slit trench is to be excavated in a humid or arid site environment. For either variation, the slit trenches are assumed to be 60 m long (surface dimension) by 6 m in depth, with a minimum 3-m spacing between slit trenches. A 1 m wide trench bottom is assumed, and also assuming that the walls slope at a 1:4 ratio, this results in a nominal surface width of 4 m. For the humid variation, a 0.3 m (1 ft) layer of sand and gravel will be spread on the floors of the slit trenches. Two standpipes will also be emplaced in each slit trench. For the arid variation, the sand and gravel layer is deleted, as are the standpipes.

Waste will be stacked within the slit trenches, implying a disposal efficiency of 0.75. Waste is furthermore assumed to be emplaced up to approximately 1 m below the ground surface, at which time the slit trench is backfilled with local soil. The slit trenches are then assumed to capped with a minimum of 1 m of soil and stabilized against erosion. (No special means are used to compact either the backfill or the cap.) For the humid variation, grass is planted while for the arid variation, a 6-inch layer of cobbles and rocks is used. For either variation, the slit trenches are located using a marker at one end of the trench, and a monument describing the trench contents at the other end.

The above considerations imply an available disposal volume per slit trench of about 985.5 m^3 for the humid version and about 1003 m^3 for the arid version. This translates to a volumetric disposal efficiency (EFF) of about 985.5/208 = 4.73 m for the humid slit trench, and about 1003/208 = 4.81 for the arid slit trench. Assuming stacked disposal at 75% efficiency, about 739.1 m^3 of waste can be disposed per humid slit trench, and 752.2 m^3 of waste can be disposed per arid slit trench. The above 3 m spacing between slit trenches also implies a surface utilization efficiency (SEF) of 208/(63 x 7) = 0.472 for both the humid and arid versions. Disposal cell thicknesses of 4.7 m and 5 m are calculated for the humid and arid versions, respectively. Disposal of 1,000,000 m^3 of waste via slit trenches furthermore implies a total disposal area (trenches plus spacing) of 1,000,000/(.75 x 4.73 x 0.47) = 596,970 m^2 (147 acres) for the humid slit trench, and 586,560 m^2 (145 acres) for the arid slit trench.

C.3.1.8 Concrete Trench

In this report, four alternative waste disposal methods are considered in detail which are based on use of concrete. These include concrete trenches, concrete slit trenches, concrete caissons, and concrete container repackaging. Above ground bunkers are also briefly addressed. The four concrete disposal methods are

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designed principally based upon waste storage facilities in current operation at various Canadian locations, although American and French experience has also been considered. These operations cover several years of successful experience.

A common concept among the five alternatives considered is that the disposal technology itself, rather than the waste form, may be used to provide structural support to barriers designed to reduce influx of infiltration into the waste. The intent is to have the concrete structures themselves meet (as a minimum) the structural stability requirements of 10 CFR 61.56 rather than relying on waste form and backfill. To the extent that further disposal cell stability can be provided using waste form and backfill (e.g., grouting), an additional bonus is gained. However, the disposal cells are assumed to be designed so that the Part 61 performance objectives can be met in any case.

This implies that compared with shallow land burial, a number of considerations must be addressed to a more significant degree. These include the potential for seismic damage to the structure, soil and foundation stability, and the potential for deterioration of the structure over time. These considerations are especially important for the above-ground bunker concept, since the bunker in this case must provide containment of the waste to a much higher degree than the below-ground alternatives. (Once the structure was breeched there would be little to mitigate dispersion of the radiocontaminants.)

A fairly extensive (and expensive) program of site study, site analysis, testing, and quality assurance is thus implied for the concrete structure alternatives. Seismic considerations important to design include probable seismic intensities in terms of accelerations, particle velocities, and wave lengths of the vibrations. Material properties or characteristics that relate to the geomechanical behavior of strata at the site include distribution of uplift pressures, bearing capacities, consolidation behavior, shear failure criteria, and the potential for liquefaction. Factors important to concrete deterioration, and which need to be considered when designing, constructing, and using the structures, include freezing and thawing, aggressive chemical exposure, abrasion, corrosion of steel and other metals embedded in the concrete, and chemical reactions of aggregates.

In any case, the reference concrete trench is assumed in this report to consist of a below-ground structure having vertical sides and constructed of reinforced concrete. Each concrete trench is assumed to contain 12 pairs of disposal cells (24 individual disposal cells) as illustrated in Figure C.6. Each disposal cell is covered by a replacable reinforced concrete lid (see Figure C.7), resulting in 24 replacable lids per trench. This practice reduces exposures to site personnel, since waste already emplaced will be covered and the concrete lid and walls will provide shielding. It also helps to reduce potential environmental releases, since possible problems developing in a given cell may be isolated and mitigating measures taken without the problems impacting the entire concrete trench.

Each disposal cell is assumed to have inside cross-sectional dimensions of about 6 m by 6 m, and a depth of about 6 m. The four disposal cells on the ends of the concrete trench are slightly smaller--measuring about 5.85 m x 6 m x 6 m deep--to enable the lids to fit properly over these end disposal cells. A slight adjustment on the sizes of the end disposal cells is judged to be preferable to manufacturing two slightly different sizes of disposal cell lids.

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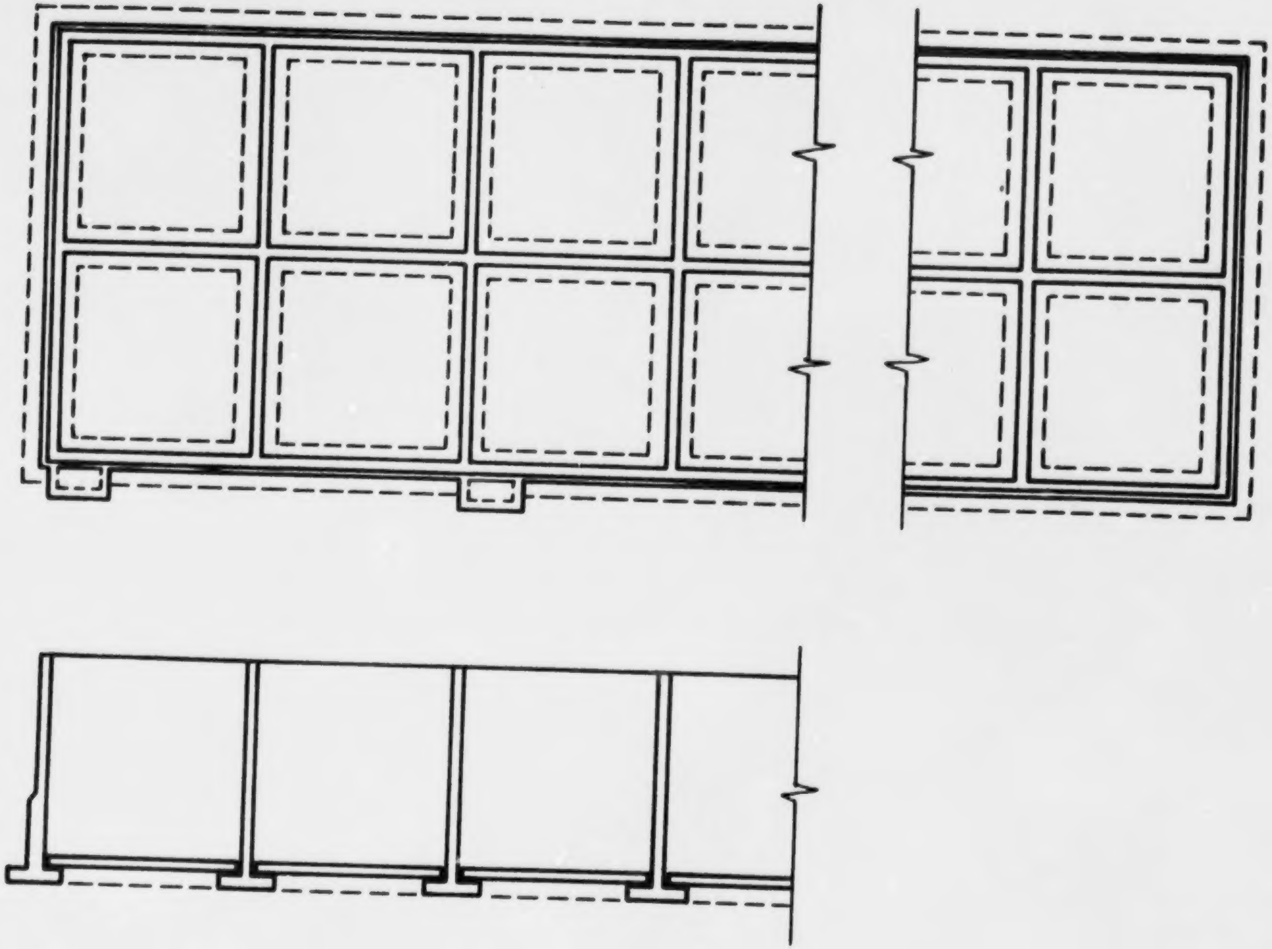
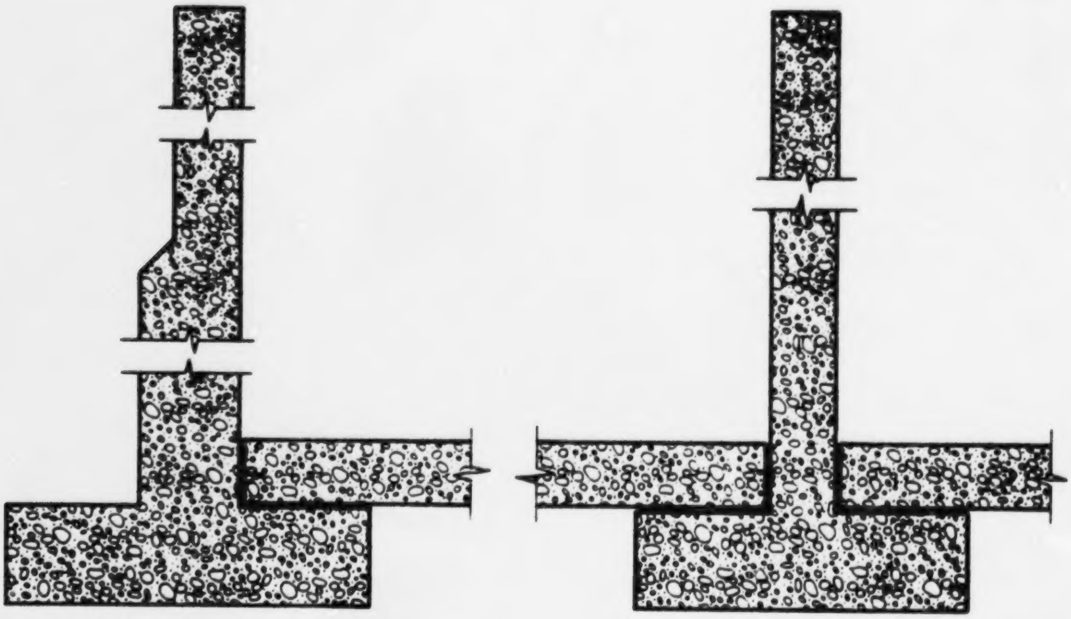
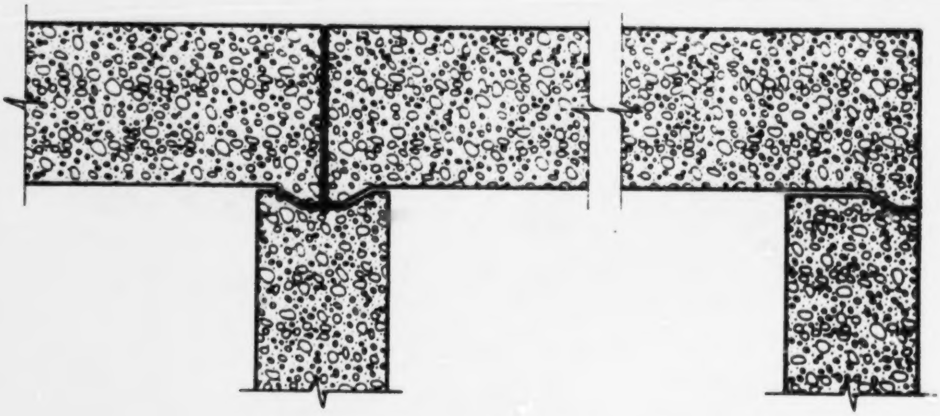


Figure C.6. Top and Side Views of Concrete Bunker



Outside Wall

Inside Wall



Lids

Figure C.7. Conceptual Details of Walls and Lids

The disposal cell lids have overall dimensions of about 6.3 m in width by about 6.45 m in length. They are designed with a protruding lip around the edges which fits into corresponding indentations built into the tops of the exterior and interior walls. The lids are fitted with lifting handles, and are designed to resist moments during lifting. The lids vary in thickness from about 0.5 m on one end to about 0.4 m on the other end. When properly installed so that the thicker ends are placed over the interior walls of the concrete trench, a slight penthouse results. This promotes runoff of percolating water from the lids.

The disposal trench is constructed in a fairly massive manner in an effort to ensure structural stability over a long time period. Walls around the perimeter of the concrete trench are assumed to be 0.3 m thick for the top 4 m and increase in thickness over the bottom 2.3 m to 0.5 m, where they are anchored in large footings. The interior walls are assumed to be 0.3 m thick for the entire 6.3 m. but also are anchored in large footings. A 0.3 m thick reinforced concrete floor is also assumed which is poured after construction of the footings.

Exterior wall footings are of simple design for three of the sides, and, for the humid design, more complex along the remaining side as depicted in Figure C.8. This is to allow construction of a series of six disposal cell sumps, where one sump is used to monitor any potential drainage from four disposal cells. (Drainage channels are built into the disposal cell floors along the walls as shown.) The disposal cell sumps are located outside of the disposal cells in order to eliminate the need for monitoring pipe penetrations of the disposal cell lids and walls. The sumps are also built directly into the footings in an effort to eliminate the potential for differential settling to separate the sumps from the concrete trench. The sumps measure about 0.6 m x 0.6 m x 1.2 m in inside dimensions, and about 0.9 m x 0.9 m x 1.8 m in outside dimensions. A standpipe is inserted in each sump.

To construct a concrete disposal trench, a large excavation is first made--i.e., sufficient to leave a 5 m space completely around the outside of the disposal trench walls--with excavation walls sloping back at a 45° angle; an exception to this is a ramp (5 m wide at the top, 30° slope) constructed at one end of the excavation. This large excavation is assumed to be 7 m deep, and within this excavation, a smaller excavation is made having rough dimensions of 15 m x 78 m x 0.5 m. This small excavation is slightly sloped to one side and one end, and two small (1 m x 1 m x 0.5 m) sumps constructed along the low side of the excavation. One sump is located at the end of the smaller excavation in the low corner while the other is located about halfway up the length. A standpipe is located in each sump. (These two sumps and standpipes are assumed to be deleted for the arid design.) The excavation is then filled with layers of gravel and sand to form a substrate, and the concrete disposal trench constructed on this substrate.

Trench construction begins with construction of the footings, followed by construction of the trench floors. Trench walls are then constructed. After the poured concrete has set up, the final forms are removed, and the outside walls are sprayed with a waterproofing material. The excavated area around the concrete walls is assumed to be backfilled with originally excavated soil. This backfill soil is compacted using heavy machinery up to a level approximately 6 inches below the top of the open trench. At humid sites, a layer of

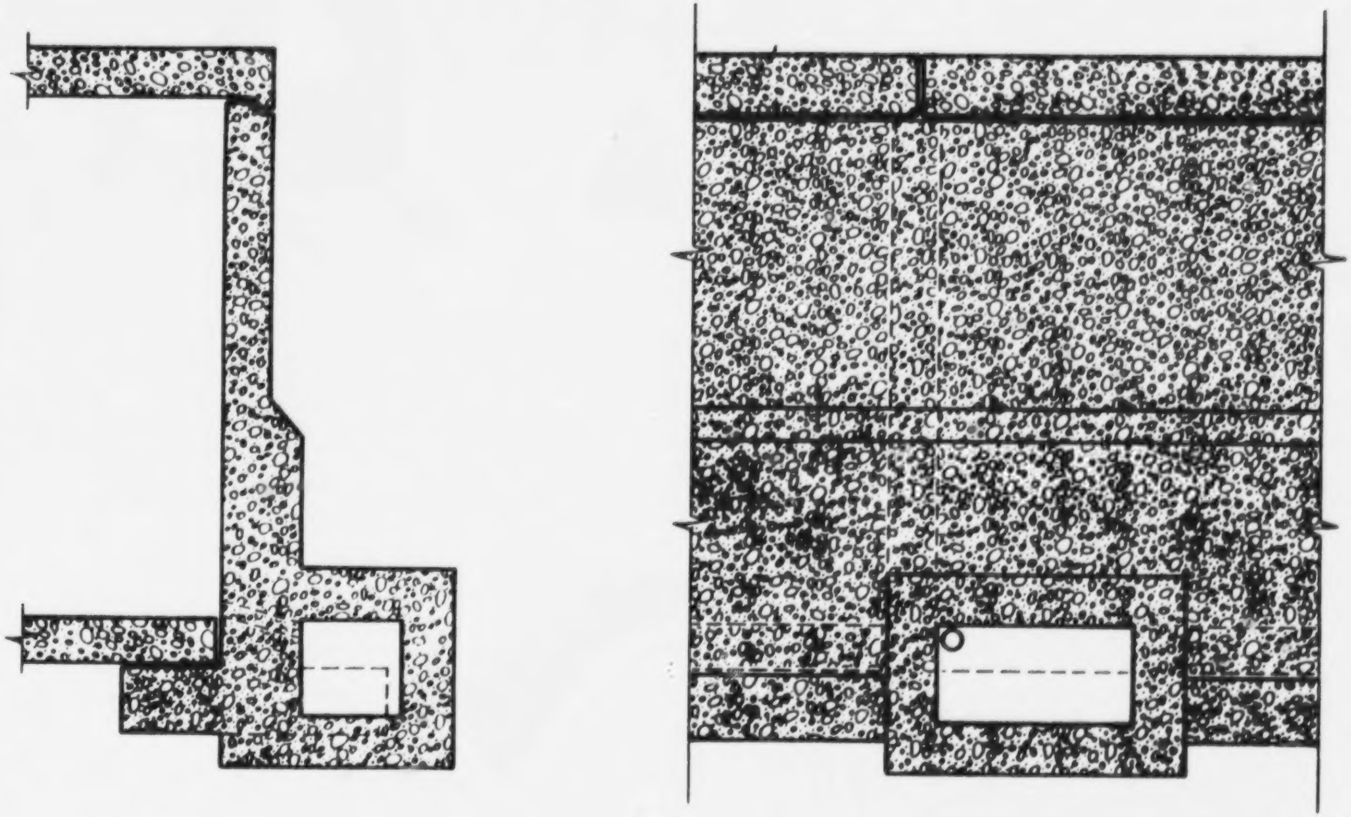


Figure C.8. Conceptual Details of Concrete Bunker Sump

asphalt is then laid down around the concrete trench and in the space between trenches. The protrusion of the trench walls above the asphalt will prevent water runoff from flowing into the open trenches; it will also help to reduce the possibility of miscellaneous debris being kicked into the open trenches. The asphalt layer is deleted for the arid design.

The asphalt layer serves several functions. First, it provides a smooth, clean, and stable working area. Second, despite efforts to compact the backfill placed around the concrete walls, the backfill will still be somewhat more permeable than unexcavated soil. This provides a preferential route for water infiltration into the soil surrounding the trench walls. At one Canadian site using below-ground bunkers, this infiltration caused problems with newly emplaced storage bunkers (Ref. 7). An asphalt layer cured the problem. Third, the asphalt layer helps to provide controlled, rapid runoff of rainwater. In this report, the asphalt is assumed to be appropriately sloped, with drainage ditches strategically located, so that water is channeled to a few points where it is monitored prior to leaving the site.

The preceding assumptions result in a total of eight standpipes per trench for the humid design and six standpipes per trench for the arid design. The standpipes end at the ground surface in prefabricated access boxes.

To dispose of waste, a 0.15 m (6 in) layer of gravel is first assumed to be placed on the bottom of the disposal cell. This allows any moisture in the disposal cell to drain away from the waste, thus avoiding waste immersion in water. Waste is then stacked within the cells up to a level about 6 inches below the top of the trench walls, after which the lid is replaced. Backfill is also emplaced; depending upon the physical and radiological characteristics of the waste, backfill is emplaced either as waste is being disposed or after the disposal cell is completely full of waste. (This latter approach increases the difficulty of filling voids between waste packages, but may be necessary for some wastes.) After the disposal trench is completely filled, the lids and exposed sides of the concrete trench are assumed to be sprayed with a waterproofing material. The disposal cells are then given a final cover as discussed in Section C.3.2.

A disposal efficiency of at least 0.75 can probably be achieved using this disposal method. Assuming a 6 inch gravel layer, this implies an available disposal space of 4,904 m^3 and a total waste disposal capacity of about 3,678 m^3 per concrete trench. At a total waste volume of 1,000,000 m^3 over 20 years, this implies construction of 272 concrete trenches, or about 14 concrete trenches per year.

The volumetric disposal efficiency (EFF) is 5.7 m, and, assuming an average 10 m spacing between each concrete trench, the surface utilization efficiency (SEF) is about 0.44. The disposal cell thickness is about 5.7 m for both the humid and arid variations.

The above assumed disposal trench design, construction, and dimensions are somewhat arbitrary, but are believed to be reasonable considering the generic nature of the project. For a real disposal site, a designer would consider such factors as the volume and radiological characteristics (e.g., surface radiation levels) of the projected daily throughput of waste. For example, a disposal cell may be sized to accept one (or a specific fraction or multiple

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of one) day's waste delivery. Larger or smaller disposal sizes might be used. The cell dimensions assumed in this report are sized to be large, yet not so large that the disposal cell lids become terribly difficult to remove and replace. The assumed cell dimensions result in a lid weight of about 40 tons, which can be handled using commercially available cranes.

Thicker walls (or other modified designs) may also be used for some wastes having high surface radiation levels. This would provide additional worker protection. It is believed, however, that such waste and site-specific considerations can best be worked out on a site-specific rather than a generic basis.

C.3.1.9 Free-Standing Above-Ground Bunker

In the past few years there has been interest in the option of using above-ground concepts for waste disposal. There may be a wide range of possible designs, some of which seem to be merely small variations of below-ground concepts such as concrete trenches. A case at one end of the spectrum would be use of free-standing structures.

This concept appears to have a number of theoretical advantages as well as a number of practical disadvantages. One advantage is that waste disposal can be theoretically carried out in a controlled, engineered, manner. In addition, the bunkers can be designed so that water possibly percolating into the disposal cells can be drained away in a controlled manner, and the drainage monitored for contamination. They can be designed so that the waste can theoretically be readily retrievable. The disposal method may also provide more of a sense of psychological well being to the public.

A few of the disadvantages include the fact that free-standing, above-ground bunkers must remain self-supporting for several hundred years. Thus, the bunkers would have to be constructed much more robustly than below-ground techniques. There would also be a lack of radiation shielding provided by soil, so different wall thicknesses would probably need to be used to reduce exposures to site personnel.

The long-term performance of the facility is also difficult to project and to model. As stated earlier, seismic considerations will be of greatly increased concern compared to other disposal technologies, as will erosional considerations. A big consideration is that unlike other disposal technologies, the site characteristics play less of a role in mitigating environmental releases. One must design for absolute containment of waste, while for other disposal technologies one is designing mostly for structural support.

It is beyond the scope of this report to perform a detailed design of a free-standing above-ground bunker disposal facility. Therefore, detailed design considerations such as the required wall thicknesses to provide shielding as a function of surface radiation levels of waste packages, will not be addressed.

For the purposes of this report, the above-ground bunker is assumed to be designed and constructed in a somewhat similar manner as the below-ground concrete trench in Section C.3.1.8. Construction would have to be more robust, however, to account for increased shielding requirements and also to provide a structure more likely to last over the long term. A substrate is first laid and then forms and reinforcing material emplaced. An underground

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drainage network is also emplaced so that any water passing through the bunker will be drained from the substrate in a controlled manner. This underground drainage network lies only a few feet under the ground and is equipped with sampling points at various locations. After the poured concrete has set and the forms removed, an asphalt layer is emplaced on the ground surface between the bunkers. This again provides a smooth, stable working area, as well as controlled drainage of surface runoff from the site. This drainage network is maintained separately from the underground drainage network and is monitored independently.

Otherwise, the above-ground bunkers would be designed and used in a similar manner as the below-ground concrete trenches. Each trench is divided into 24 separate cells covered by 24 replaceable lids. Each cell has a layer of gravel on the concrete bottom. Waste is emplaced using cranes and, after disposal, the exposed concrete is assumed to be weatherproofed.

C.3.1.10 Concrete Slit Trench

This alternative is very similar to the below-ground concrete trench alternative considered in Section C.3.1.8. One major difference is that the concrete slit trenches are much smaller in size. This disposal alternative may be viable when there is only a small amount of waste to be disposed using concrete bunkers. As before, there are two slightly different versions depending upon whether the disposal technology will be located in a humid or an arid environment. A humid design is implicitly assumed, while arid differences are noted in the following discussion as appropriate.

An illustration of this disposal concept is provided as Figure C.9. Each concrete slit trench is assumed to have gross surface dimensions of about 2.6 m x 20.6 m, and is also assumed to be divided into five individual disposal cells having inside dimensions of 2 m wide by 3.75 m long by 6 m deep. Each of the individual disposal cells is assumed to be covered by a replaceable concrete lid, and each lid has a slight penthouse shape to promote runoff of water. Due to the relatively small sizes of the disposal cells and lids, construction is assumed to be somewhat lighter than that for the concrete trench.

To construct the trench, an excavation is first made sufficient to construct six concrete slit trenches at once. For safety, excavation walls are assumed to slope back at a 45° angle, and a 5 m wide ramp is constructed at one end for vehicle access. A 0.3 m concrete slab is then laid for each slit trench on top of a gravel and sand substrate. The slab slopes slightly from side to side and from end to end, and a concrete drain runs the length of the trench along the side of the trench. A 0.6 m x 0.6 m concrete sump is also built into the side of the trench which extends from the top of the trench to a level 0.6 m below the bottom of the concrete slab. This bottom 0.6 m is actually used as the sump and, after a standpipe is located, the remaining 6 m is grouted. The standpipe ends in a covered access box. A standpipe is also located in the substrate near the low end of each slit trench, and this standpipe also ends in a precast access box. (Standpipes into the substrate layers are assumed to be deleted for the arid design.)

The above design is a bit complicated; an alternative, simpler, design could involve provision of access ports in the individual disposal cell lids. The access ports would correspond to the locations of standpipes set into the corners of the disposal cells in a manner similar to that discussed earlier for the

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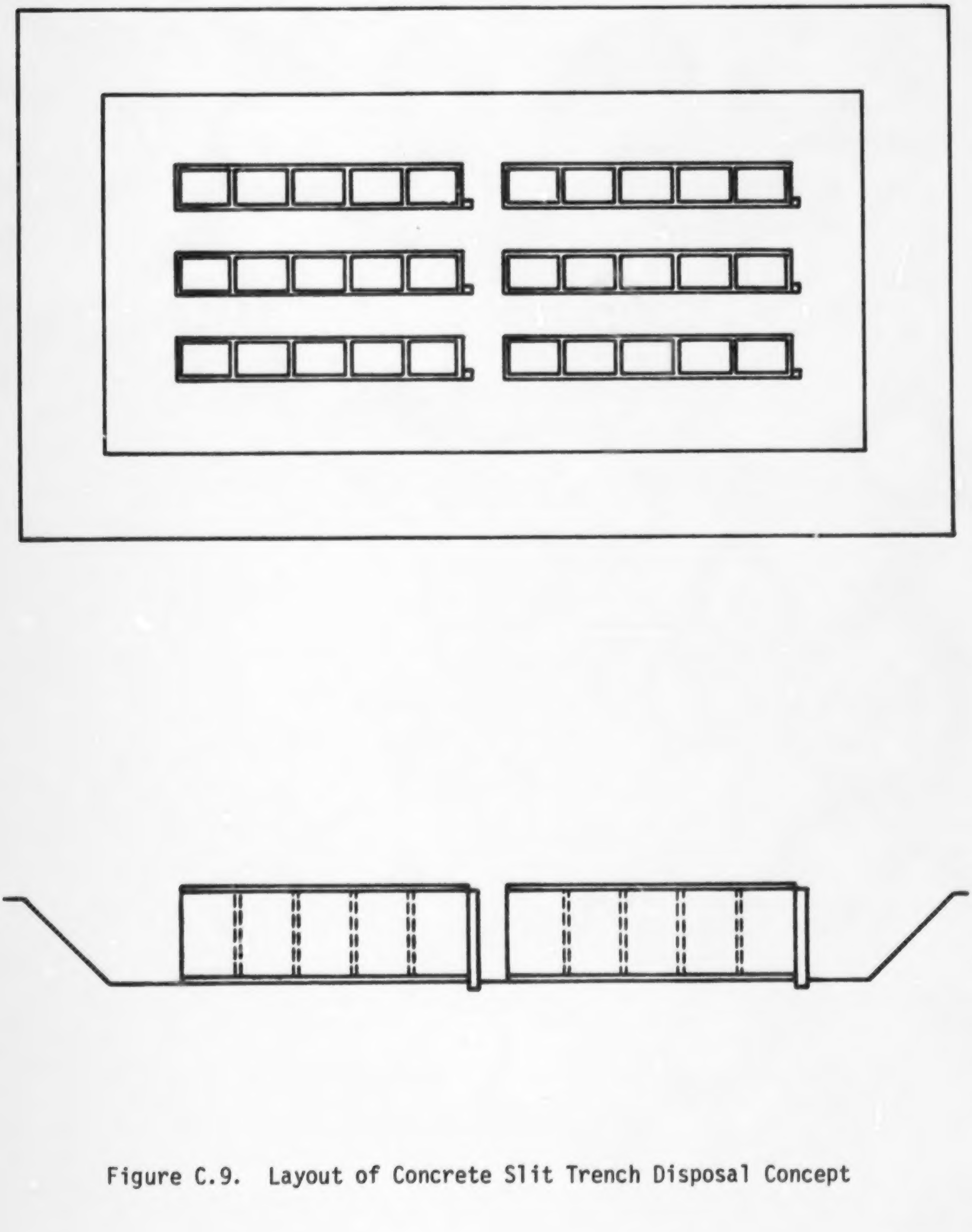

Figure C.9. Layout of Concrete Slit Trench Disposal Concept

concrete trench. This approach, or a variation thereof, has been used before in bunkers used for waste storage, but there is some question regarding the long-term performance. The essential problem is that this approach introduces a penetration in the lids, which presents a possible concentrated source of infiltration over the long term. Thus, the more complicated (expensive) approach is taken in an effort to eliminate such penetrations.

The concrete floor of each cell is covered with a 0.15 m (6 in) layer of gravel. After backfilling around the concrete up to a level approximately 6 inches below the tops of the slit trench walls, the compacted ground surface is covered with a layer of asphalt. Similar to the concrete trench alternatives, this asphalt layer is sloped and shaped so that surface runoff is drained in a controlled manner and monitored prior to release from the site. (The asphalt layer is deleted for the arid variation.) In operation, the waste is emplaced up to about 6 inches below the level of the ground surface, resulting in a disposal cell thickness of about 5.7 m. After waste is emplaced and the disposal cell backfilled, the lid is replaced and the exposed concrete surfaces are waterproofed.

As discussed earlier, the concrete slit trenches are assumed to be constructed in groups of six, and each group is constructed in a checkerboard pattern in a manner similar to that for auger hole groups. Slit trenches are assumed to be positioned 3 m from one another in each group, and the spacing between each group is assumed to be 10 m. Significant disposal parameters for this disposal technology are listed below for each disposal cell group, where these parameters have been defined previously.

Parameter	Value
Available Disposal Volume (m^3)	1283
Actual Disposal Volume (m^3) (EMP=0.75)	962
Vol. Disposal Efficiency (m)	5.7
Surface Utilization Efficiency (for disposal cell group)	0.17
Disposal Cell Thickness (m)	5.7
Disposal Cell Depth (m)	2
Surf. Area (1,000,000 m^3 of waste) (m^2)	1.34E+6
(acres)	332

Assuming a disposal efficiency of 0.75, about 31.1 m^3 of waste could be disposed per individual waste cell, or about 160.3 m^3 per concrete slit trench. At a site disposing of 1,000,000 m^3 of waste over 20 years, this results in 6238 slit trenches being constructed and used, or about 312 per year.

C.3.1.11 Concrete Caissons

Caissons have been extensively used in Canada to store low-level waste pending a future decision on waste disposal. Caissons have also been used, although significantly less frequently, in this country as both waste storage and waste disposal mechanisms. Past designs have often been based on the use of ordinary concrete culverts. Concrete culverts have most utility for regular shapes such as 55-gallon or 80-gallon drums, since culverts are readily available in a number of lengths and diameters. Diameters can be chosen so that there is little wasted space between the waste and the inside wall of the culvert. Culverts would appear to have much less utility for large bulky items such as boxes.

Another Canadian waste storage design involves use of steel reinforced cylindrical concrete caissons that are constructed in place. This design, in current use at Chalk River National Laboratory, has been developed as an alternative to use of retangular bunkers. These cylindrical caissons measure approximately 6.1 m diameter by 3.7 m deep (Ref. 7).

In this report, caissons are assumed to be constructed of steel reinforced concrete, and are also assumed to be cast in place at the disposal facility. Another option would be to purchase precast culverts from an offsite manufacturer. The approach taken by an actual licensee would depend upon a number of considerations, including the site-specific availability of manufacturers and eventual regulatory criteria on testing, materials chemical content, and quality control. Such criteria has not been developed.

For this alternative, caissons are assumed to be constructed which are about 6 m in length (including a 0.3 m base) and on the order of 0.6 m in inside diameter (Figure C.10). This leaves a remaining height within a caisson of about 5.7 m. A standard 55-gallon drum has a diameter of about 56 cm and an approximate height of 86.4 cm, which means that six 55-gallon drums can be stacked up to a height of about 5.2 m inside the caisson (or about 1.28 m^3 of waste per caisson). The caissons are assumed to be positioned vertically in the ground in groups of about 40 (e.g., a 5 x 8 array). The spacing between the caissons is assumed to be approximately 2 m.

To construct a caisson group, a large (about 6 m deep) excavation is assumed to be made. The floor of the excavation has a slight slope from end to end and from side to side. A gravel drain runs along the low side of the excavation, and a 1 ft layer of sand is emplaced on the excavation floor. The caissons are then constructed upright in the excavation using prefabricated, reusable forms, and temporarily braced. The excavation is then backfilled with compacted earth, and an asphalt layer emplaced so that the tops of the open caissons protrude about 6 inches above the asphalt. Two standpipes are located in the gravel drain. Each caisson group is also assumed to be separated from other caisson groups by an average 5-meter spacing.

In practice, waste will be emplaced within the caissons, backfilled, and capped with a concrete plug. The plug extends about 0.5 m into the caisson to provide shielding. The exposed portions of the caisson and plug are sealed with a weatherproofing material. Each caisson group is marked with a monument that identifies the caisson group and describes the caisson contents.

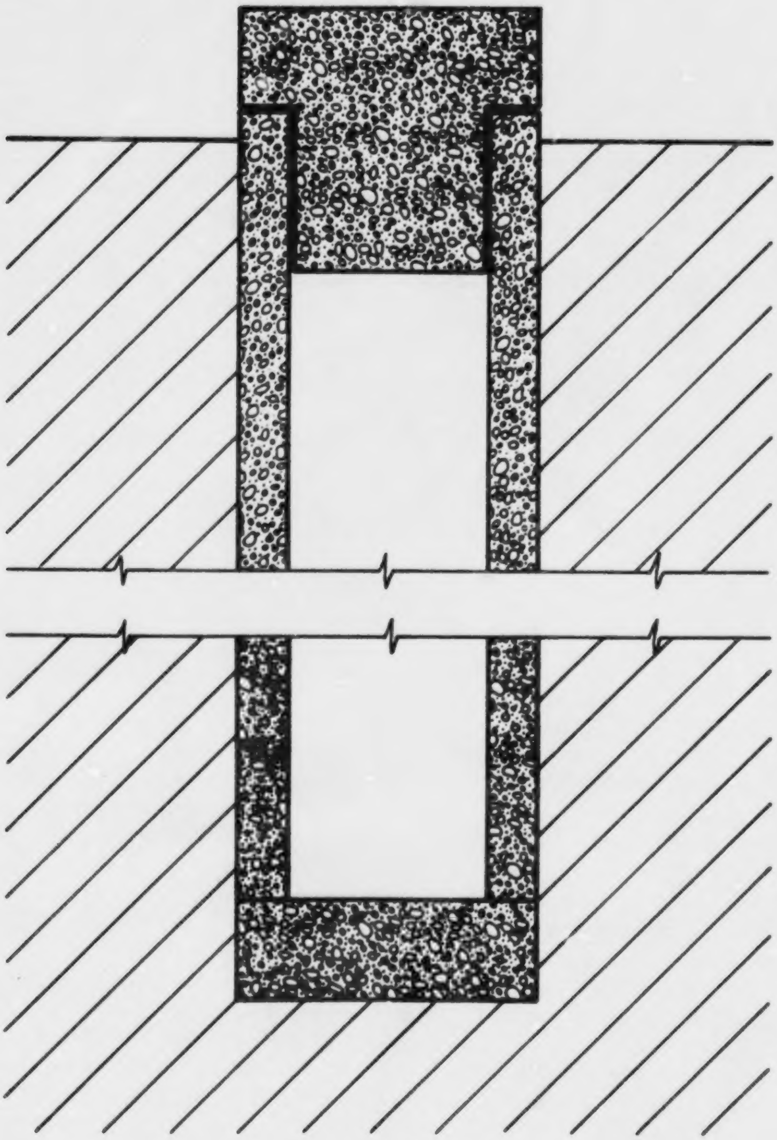

Figure C.10. Caisson Disposal Concept

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With the above assumptions, about 51.1 m^3 of waste in 55-gallon drums can be disposed in each caisson group. This corresponds, for a site accepting 1,000,000 m^3 of waste over 20 years, to construction of 977 caisson groups per year (39,063 caissons). This would obviously be too expensive for routine use. It is also unreasonable since much of the waste will be in larger packages than 55-gallon drums (e.g., 150 ft^3 liners or 128 ft^3 boxes). However, different caisson sizes may be used for different waste forms.

The above assumptions also imply the following parameter values per caisson group (40 individual caissons/group):

Parameter	Value
Avail. Disp. Vol. (m^3)	58.81
Act. Disp. Vol. (m^3)	51.1
Waste Emp. Eff. (EFF)	.87
Vol. Disp. Eff. (m)	5.2
Surf. Util. Eff.	0.025
Disposal Cell Thickness (m)	5.2
Disposal Cell Depth (m)	2
Surface Area (1,000,000 m^3 of waste)	
(m^2)	8.86E+6
(acres)	2190

C.3.1.12 Repackaged Disposal Operation

Over the past few years, the use of containers providing structural support for high-activity low-level waste forms has become quite prevalent. Current examples are called high integrity containers, are frequently constructed of high density polyethylene, and are often used by utilities and others to package resins, filter media, and cartridge filters for delivery to a disposal site.

A somewhat similar option could be considered in which waste arriving at a disposal facility is repackaged, at the disposal facility prior to disposal, into containers providing containment and/or structural support. Providing that the volume of waste subjected to such repackaging operations is small--e.g., less than perhaps a few hundred m^3/yr--repackaging activities could be performed with few modifications to disposal facility operations. The reference disposal facility design includes a waste activities building (see Section C.2.2.2) in which occasional waste solidification and repackaging (or overpacking) activities may be carried out.

The focus of this alternative in this section is for extensive use of this repackaging operation. The idea is to ensure that all waste delivered to the disposal site is eventually disposed into containers of uniform dimensions, thus enabling regular stacking of the waste containers and providing mechanical stability equivalent to or exceeding the stability criteria in 10 CFR 61.56. This disposal concept originated in France, where it has formed the principal low-level waste disposal philosophy for several years, and has been expanded by Westinghouse and developed into the Westinghouse SUREPAK concept (Ref. 8). There could be a number of container designs and stacking arrangements as shown in Figure C.11, involving such forms as cylinders, hexagonal prisms, or square or rectangular boxes. The important features of the concept, however, include

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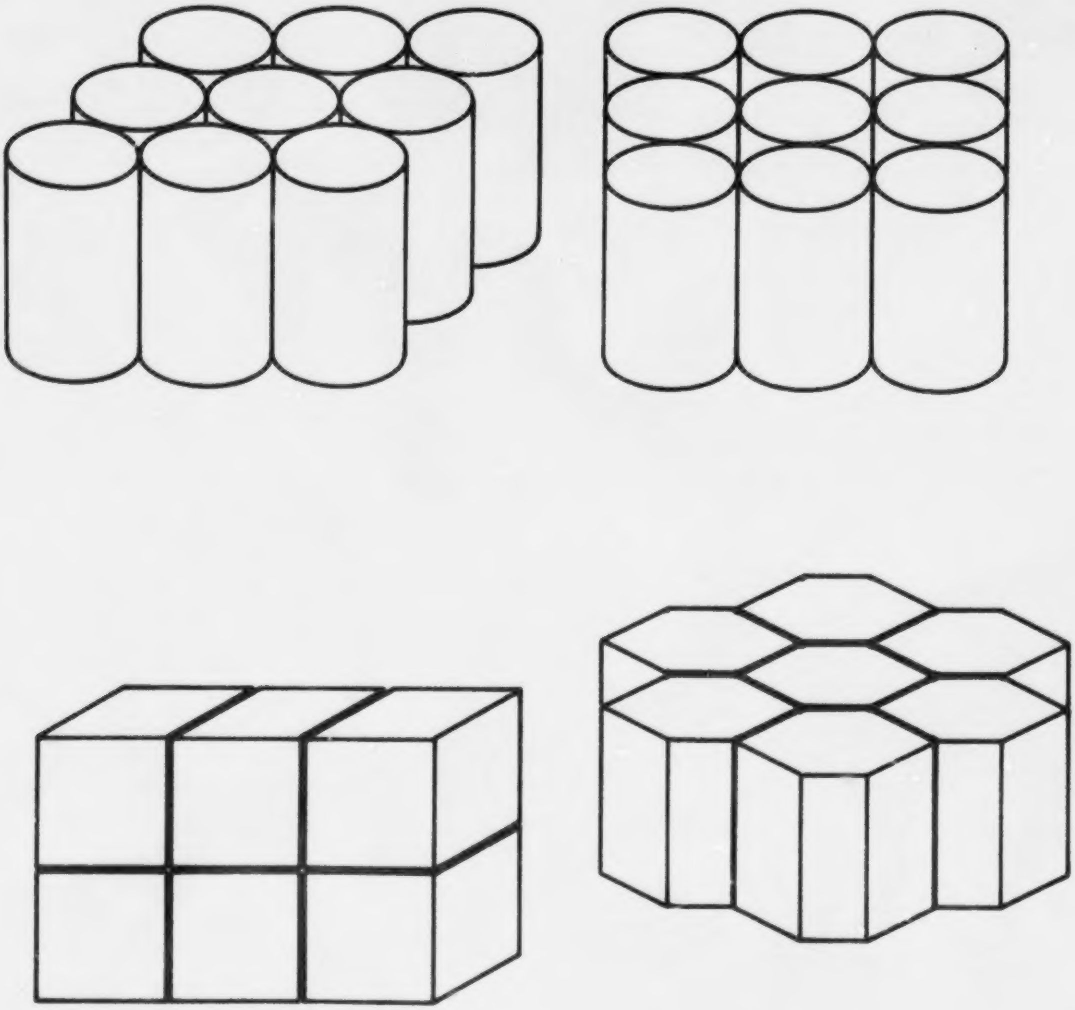

Figure C.11. Conceptual Stacking Arrangements

(1) design for container structural integrity over several hundred years, and (2) uniformity of sizes so that a regular stacking arrangement may be achieved.

Containers

It is not the purpose of this report to analyze alternative sizes and shapes of containers, nor to recommend the "best" size and shape. There may be a number of different shapes and sizes that could be used at the same disposal facility, provided that the different sizes are segregated to ensure a good stacking arrangement. The French at la Centre de la Manche, for example, use cylinders as well as rectangular boxes (Ref. 9). The purposes of this report, however, require an estimate of costs, land use, radiological impacts, etc., and to do so, a number of assumptions need to be made, including assumptions on container dimensions and processing options.

For the purposes outlined above, reference high strength structural containers are assumed which are modeled after the French "blocs"--i.e., right circular cylinders--except that a somewhat larger size is assumed. Each bloc is assumed to be constructed of reinforced concrete and have outside dimensions measuring about 2 m in diameter by 2.25 m high. The blocs have slots cast into the bottoms to enable handling by fork lifts, and also are equipped with attachment points to enable waste stacking using cranes. The available volume within a bloc is about 5.5 m^3 (~200 ft^3), into which varying volumes of waste may be emplaced depending upon the dimensions of the waste packages placed into it. This size will accommodate, for example, 14 55-gallon drums or a single 6 ft x 6 ft liner or high integrity container. An average waste packing efficiency of about 75% appears to be reasonable within the bloc, resulting in an approximate disposed waste volume per bloc of about 4.1 m^3 (~150 ft^3). For very high activity wastes, thicker bloc walls could be used. This would provide shielding to workers, but would reduce the volume that could be disposed in the bloc.

After a given bloc is filled, the remaining space is filled with grout. A reinforced concrete lid is fitted into the bloc in order to augment structural strength. Like the bloc, the lid is mass produced (one size), and fitted with prongs on the bottom which penetrate into the grout. When the grout hardens, the lid will be attached firmly. The cover of the lid has an identification number etched into it to allow each bloc to be identified individually as to waste contents. The familiar three-pronged symbol for radiation is also etched into the lid. A completed bloc would weigh about 20 tons and have an overall density of about 130-140 lbs/ft^3.

This report envisions that most such repack operations will be carried out at the disposal site, although there is no reason why waste blocs could not be filled at the generator site and shipped to the disposal site ready for disposal. An obvious disadvantage is the incremental costs for the increased mass and volume of material that must be transported. The waste generator would have to compare this cost against the cost for repackaging that would be assessed by the disposal site operator. An alternative approach (saving transportation costs) would be for waste to be packaged in a special lightweight container at the waste generator site, and then merely filled with grout at the disposal site. A special container is therefore assumed to be developed which is conceptually similar to an existing disposal bin used in France (Ref. 9). A square box shape is assumed, but the concept would work as well, if not better, in another shape such as a rectangular box or a cylinder. The bin is assumed to be 2.25 m square in outside dimensions.

This approach would be of most use for contaminated piping, machinery, or other forms of noncompactible waste. In this case, a metal (carbon steel) bin would be used which has the same outside height as the concrete bloc. A perforated metal basket is hung within the bin leaving several inches of space between the basket and the bin walls and floor. Reinforcing bars protrude from the inside walls of the bin. The waste generator would fill the basket with loose waste, fasten on a lid, and ship the bin to the disposal site. At the disposal site, cement would be pressure-injected into the void space between the bin and the basket. The cement would also flow into and fill the voids between the individual waste components. After the cement had set, the bin would be disposed in a similar manner as the blocs.

Site Processing and Repackaging Operations

The assumed waste processing and repackaging operations are based on current ongoing operations at la Centre de la Manche as well as an approach proposed by Westinghouse (Refs. 8, 9). For this report, all waste processing and repackaging activities are assumed to be carried out in a large building. This structure, called the waste process and repack building (WPRB), is in addition to the six structures assumed for the reference site. The central features of the WPRB include separate waste repackaging operations for low- and high-surface radiation level waste containers (wastes that can and cannot be contact handled), a nondestructive analysis station for waste verification, and temporary storage areas for both incoming and repackaged waste.

In operation, waste would be first checked into the site in the usual way. The waste transport vehicle and cask (if any) would be checked for fixed and removable contamination, as would occasional waste packages which can be contact handled. The manifest documents would be recorded as would the waste classification. Depending upon the waste package surface radiation level, the waste transport vehicle would be directed to either the contact or remote handling area of the WPRB. At the remote handling area, waste containers are transferred directly into the blocs using a crane and a shielded transfer bell. A storage area is available for filled and grouted blocs. A small area behind a shield wall is also available for temporary storage of incoming irradiating wastes.

At the contact handling area, further waste segregation takes place. As an option, some of the waste may be reduced in volume prior to placement into a bloc. (A number of volume reduction options are considered in this report involving operation of an adjunct processing center to the disposal facility. Depending on the waste spectra (see Appendix B), compressible waste may be either compacted or incinerated, and noncompactible waste may be compressed using a large hydraulic press.) Noncompactible wastes such as liquid scintillation vial waste and animal carcasses are generally placed directly into blocs and grouted. Wastes shipped to the site within blocs or bins are grouted. As before, small temporary storage areas are available for incoming waste containers as well as filled blocs and bins.

Arriving waste packages are periodically selected on a random basis for nondestructive analysis. Within the analysis room, a waste container is scanned for gamma emitters using scintillation detectors. This is to confirm the radionuclide description and quantity as stated on the shipment manifest. An effort is also made to confirm the physical description of the waste through measurement techniques such as X-rays, weight, and ultrasound. This would be done as a precursor to the optional volume reduction activities.

To obtain maximum compaction efficiency, the waste containers being compacted should contain only a minimum amount of noncompactible material, and liquids would also be excluded for safety reasons. Thus, waste acceptance criteria for compaction operations would specify a maximum amount of noncompactible material in a container, and also would exclude liquids such as scintillation vials. The confirmatory analysis is needed to ensure that waste generators will comply with the waste acceptance criteria. Despite this, however, the compactor would need to be equipped with catchment systems for any liquids that might be in a package in violation of the criteria. Procedures would be in place to process this liquid prior to disposal.

Waste Disposal

Design of waste disposal operations involves a number of considerations. For example, waste disposal could be carried out below-grade, partially above-grade, or fully above-grade. The French disposal system at la Centre de La Manche is constructed about half above and half below grade, leaving a large, earth covered mound. These operations, however, result in stacked waste being exposed to the elements for up to several years prior to capping. This results in a considerable quantity of water being collected in the facility drains. This water has the potential for being contaminated, and so current French practice is to pipe the collected water over to an adjacent reprocessing plant for treatment. Treatment generally consists of discharge to the English Channel (Ref. 10). A below grade approach allows for capping operations to closely follow waste disposal.

Another consideration is that to avoid the potential for differential settling of stacked waste containers, plus allow for precise location of containers, it would be preferable to stack the waste on a solid foundation such as a thick concrete slab. In a humid environment, such a solid surface would result in a considerable quantity of runoff water which could be contaminated. Facility operational policy procedures would probably require that this water be collected and monitored for processing or release. This implies that additional attention should be paid to water control measures such as drains and sumps. In so doing, one is advised not to place the location of such drains and sumps too close to the water table.

Thus, one possible design in a humid environment would be to design the facility for completely controlled gravity drainage to a centralized catchment basin. In addition to the need to incorporate sufficient space to the water table to emplace drainage structures and pipes, this implies the need for a site with sufficient topographical relief to allow gravity drainage.

For this report, two general disposal designs are assumed: one humid and one arid. The humid design is constructed so that about one-third extends above the original grade. For the reference disposal sites this is believed to be sufficient to allow for proper drainage yet not far enough above grade so that protection against such potential problems as erosion cannot be readily accommodated using standard engineering techniques. The arid design is assumed to be entirely below grade. Each is individually discussed below.

Humid Design. Waste disposal operations are conceptually illustrated in Figure C.12. A large trench is first constructed, having a slight slope in the floor from side to side and from end to end. The dimensions of this trench

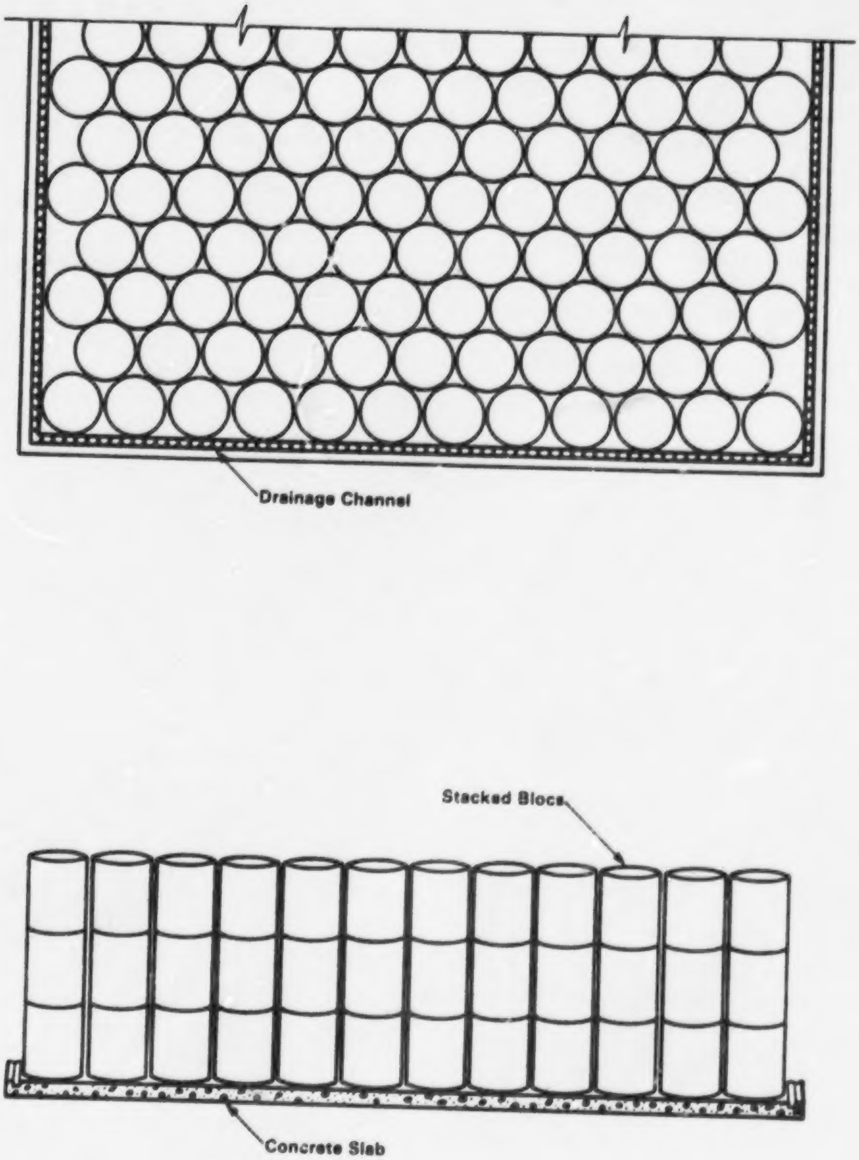
Drainage Channel
Stacked Blocs
Concrete Slab

Figure C.12. Repack Disposal Concept

are 30 m by 180 m on the surface, with a depth of about 5 m. The trench walls slope at a 1:4 ratio, and a 30° ramp is built into one end. The walls are shotcreted to reduce the potential for slumping. A 0.5 m deep excavation is made for a 0.5 m-thick gravel and sand substrate which is emplaced and compacted in layers. This is followed by a 25 m x 170 m by 0.3 m-thick reinforced concrete slab, which provides a solid base on which to stack the blocs and bins. A drainage network is built into the slab, in this case running around the perimeter of the slab and also bisecting the slab lengthwise. The drainage network ends in a 4 m x 4 m by 2 m deep sump as shown.

The filled blocs and bins are stacked into place using cranes and forklifts, and the trench and stacked waste are backfilled and capped as operations progress. Soil is assumed to be used to backfill around the sides of the stacked waste. However, the code user has an option to consider different types of backfill between the stacked blocs (see Table 2-16). Gravel is used as a backfill at la Centre de La Manche, and according to French officials, the optimum size for this gravel backfill seems to be about 2-8 mm in diameter (Ref. 10). A standpipe is placed into the sump which terminates at the site surface in a standpipe access box.

Although the above approach should be adequate for most wastes, a need will occasionally arise to dispose of oversized wastes such as control rods or large components. In this case, special oversize waste disposal containers are assumed to be used which have dimensions which are integer multiples of the dimensions of the reference containers. An example would be a bloc having the same diameter but a height three times the height of a standard bloc. This 6.75 m length would be more than sufficient for reactor control rods.

The above assumptions imply that the blocs and bins would be stacked three high (disposal cell thickness = 6.75 m; disposal cell depth = 2 m). Assuming that most of the waste is disposed in blocs (e.g., about 90%), about 22,000 m^3 of final waste container may be disposed per trench. The above assumptions also imply a total waste capacity per trench of 12,500 m^3, resulting in an overall waste emplacement efficiency across the solidified mass of about 43.7%. This is calculated as the total waste volume (12,500 m^3) divided by the total theoretical volume above the slab (i.e., 6.75 m x 25 m x 170 m). This further implies a volumetric disposal efficiency of 6.75 m. Assuming a 10 m spacing between slabs, this also results in a surface utilization efficiency of about 67.5%. Assuming 1,000,000 m^3 of waste over 20 years operation, about 80 reference disposal trenches would be constructed, or about 4 per year.

Arid Design. This design is very similar to the humid design except that less emphasis is placed on water control. The trench excavation is somewhat deeper -- 8 m rather than 6 m -- and so waste disposal takes place entirely below grade. A sump is installed to provide for water control during operations. Once the disposal trench is closed, there is assumed to be little need for below-grade water control, and so no standpipes or access boxes are assumed to be installed. The standard arid disposal cell cover is also assumed to be emplaced which incorporates rocks rather than grass for erosion control.

It may also be noted that at an arid site, disposal trenches may be constructed at much greater depths -- e.g., perhaps at 1½ times or twice the assumed depth. There is no reason why the depth of disposal could not be greater at an arid site. However, eventually a safety limit will be arrived at for waste stacking. In addition, the slab, blocs, and bins would have to be designed to withstand a larger compressive weight. Finally, the greater depths of excavation imply greater concerns for worker safety. Unlike the earlier trench variation B, considerable time must be spent at the bottom of the trench in order to construct the slab and to stack the waste containers. The excavation walls would probably have to be either shored or laid back at a gentler slope than the assumed 1:4 ratio. The latter case would imply a need for cranes having longer booms. More ground space would also be needed to construct an entrance ramp.

C.3.2 Alternative Disposal Cell Covers

This section describes the alternatives considered in this report for covering (capping) disposal cells. A properly installed disposal cell cover will serve several functions, including helping to minimize erosion, providing a radiation shield, and forming a barrier to infiltration of rain or surface water. Appendix E of the draft Part 61 EIS (Ref. 1) provides a review of several considerations important to installation of disposal cell covers, as well as long-term performance of the covers (caps).

In this section, two basic cover alternatives are considered. Sections C.3.2.1 and C.3.2.2 discuss these alternatives in terms of waste disposal cell alternatives not involving the use of concrete. (These alternatives include the reference disposal facility and trench variations, the slit trench, and the auger hole variations.) The cover alternatives include a base or reference design and an improved design. The base and improved designs each have a variation appropriate for a humid and an arid environment. Section C.3.2.3 addresses the above for disposal cell alternatives involving the extensive use of concrete.

C.3.2.1 Reference

This cover alternative has been frequently referred to in Section C.3.1. Essentially, the cover consists of a minimum of 1 m of rather indifferently compacted soil. This soil is available local to the disposal site, and much if not all consists of soil originally excavated from the disposal cells. Thicker layers of soil may also be used here and there on the disposal site to help promote drainage.

In the humid variation, the cover is assumed to be stabilized against erosion by planting a short-rooted form of vegetation such as grass. In the arid variation, however, the grass cover is assumed to be replaced by a 6-inch layer of cobbles and rocks.

C.3.2.2 Improved

This alternative provides for greater structural stability of the disposal cell covers and, for the humid variation, also provides an improved barrier against moisture infiltration.

In the humid variation, the cap is again assumed to have a minimum thickness of 1 m. In this case, however, the cap consists of a 0.5 m layer (minimum) of compacted clay followed by a 0.4 m layer (minimum) of compacted soil and a 0.1 m layer of topsoil. (The clay is assumed to be obtained from an offsite source.)

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The clay and soil layers are laid down in successive 8- to 12-inch sheets and are compacted using heavy machinery. Following emplacement and compaction of the clay and soil layers, a grass cover is planted.

In the arid variation, the cap is again assumed to consist of a minimum 1 m of local soil. However, the soil layer is assumed to be compacted in successive 8- to 12-inch layers using heavy machinery such as a vibratory compactor. In addition, the cover layer of rocks and cobbles is increased from a 6-in thickness to a 12-inch thickness.

The above assumptions represent only two examples of a range of possible improved--or at least more expensive--disposal cell covers. Another alternative could consist of a disposal cell cover of materials such as alternative layers of clay, gravel, and/or sand in order to produce a "wick" effect. Additional whistles and bells could be provided by layers of synthetic membranes. Use of such membranes is recommended by EPA for hazardous waste landfill disposal facilities.

Such elaborate designs are not considered in detail in this section for a number of reasons. One reason is that there may be a large number of such variations. Another reason is that a generic approach is taken in this report, and that consideration of more detailed designs is believed to be more appropriate for a specific site. A third reason is that while the long-term performance of engineered materials such as synthetic membranes is difficult to predict, there is much more confidence regarding the longevity of natural materials such as clay. Finally, it must be recognized that the usefulness of more elaborate cover designs is entirely a function of the care with which the cover is emplaced, and of the structural stability of the disposal cell. Disposal cell covers, for example, are generally applied using bulldozers, which imply that relatively simple designs are the only designs likely to be properly emplaced. In addition, the best design on paper will count for little in the event of settling and subsidence of disposal cell contents. Such settling will almost certainly disrupt any wick effect, and some settling will probably occur even in disposal cells containing stabilized waste.

C.3.2.3 Concrete Disposal Alternatives

In Section C.3.1, three disposal alternatives were considered which involved below-ground disposal of waste in concrete structures. These include concrete trenches, concrete slit trenches, and concrete caissons. All of these alternatives were designed so that water infiltration would be minimized and that runoff could be controlled and monitored. The walls and lids covering the trenches and caissons protrude slightly above the ground surface, which is covered with an asphalt layer (humid variations only). The asphalt is sloped and features controlled drainage, where runoff is channeled through shallow ditches to a few points where it can be monitored prior to release from the site. The lids and concrete walls are sealed and waterproofed. Thus, the appearance of these concrete disposal variations (prior to covering) is that of a series of concrete protrusions above an earthen or asphalt surface.

For the repackaging option, no asphalt layer is assumed for either the humid or arid variations. However, the humid version is assumed to be designed so that slightly less than 1/3 of the disposed waste volume protrudes above the original ground surface. (For the arid version, all waste is disposed below the original grade.) After backfilling around the sides of the stacked waste, the appearance (prior to covering) will again be that of a large concrete mass protruding above the ground surface.

A State or Federal licensing agent may prefer that the disposal site be left in this condition for several years following closure of the site. The performance of the site could be closely monitored, and if any potential problems were identified with any disposal cell or structure, the waste could be potentially retrieved. This option is not considered at this time, however, since the increased complexity is beyond the scope of this project.

In this report, therefore, the concrete disposal cells are assumed to be given their final cover soon after the waste is emplaced. This section describes the base and improved designs as specifically implemented for the concrete disposal alternatives. Two variations for each cover design level are considered: one for a humid site and one for an arid site.

Base Alternative. In both the humid and arid variations, a layer of on-site soil is laid over the asphalt thick enough to extend 1.5 m above the tops of the disposal lids. The spaces between the individual disposal cells are filled in with soil to leave a relatively flat final surface lying approximately 2 m above the original ground surface. (For the humid repackage option, the final surface is approximately 4 m above the original grade.) A grass cover is then planted for the humid variation, while a 6-in layer of rocks is emplaced in the arid variation.

Improved Alternative. Three of the humid disposal alternatives are characterized by a series of concrete lids protruding above an asphalt surface (below-ground bunker, concrete slit trench, caissons). For these disposal technologies, the improved cover alternative consists of a 0.3 m layer of gravel followed by a 0.3 m layer of sand. This sand layer covers the lids and extends about 6 in above the top of the lids. This is followed by a 0.5 m clay layer, a 1.25 m layer of soil, and a 0.1 m layer of topsoil planted with grass. All layers are compacted in thin sheets using appropriate construction equipment.

This design is adopted due to the assumed structural stability of the concrete disposal cells as well as the existence of the asphalt surface. The original drainage mechanism over the asphalt surface is maintained so that water percolating through the improved cover can be monitored for volume and activity. The subsurface drain will also aid in determining a percolation source term through the final cover and thus aid in predictions of potential impacts over the long term.

For the arid versions of the above, the spaces between the disposal cells are filled with compacted soil up to the approximate top of the waste (or disposal cell lids). A 2-m thick layer of soil is then installed which is compacted in sheets. This is finally followed by a 12-in layer of rocks.

For the humid repackage alternative, the space between the disposal cells is assumed to be filled with compacted soil up to the top of the disposed waste. A cap is then installed similar to that discussed above -- i.e., a 0.3 m gravel layer, a 0.3 m sand layer, a 0.5 m compacted clay layer, 0.8 m of soil, and 0.1 m of topsoil. The arid version is assumed to be disposed below-grade in a similar manner as the reference trench. The disposal cap is then constructed so that it lies 1 m above original grade and is stabilized using a 12-in layer of rocks.

The monuments are relocated to the earth's surface. In addition, the monitoring wells associated with each trench or caisson group are extended to the earth's surface. The well standpipes terminate as before in sampling boxes located flush with the earth's surface.

C.3.3 Waste and Backfill Compaction Alternatives

In Section C.3.1, several disposal alternatives were considered in which waste was assumed to be emplaced into the disposal cells, and soil backfilled up to the level of the earth's surface. Little or no compaction was applied to the backfill, which will result in a certain amount of consolidation over time. For example, voids would be expected in the interstitial spaces between the waste packages.

These alternatives are intended to help consolidate the backfill and work more of it down into the interstitial spaces between waste packages, thus reducing the voids within the disposal cells. The alternatives considered in this section include measures taken to compact the waste and backfill prior to installation of the disposal cell covers. Thus, measures considered in Section C.3.2 to compact the disposal cell covers are independent of the measures considered in this section to compact the waste and backfill. Three alternatives are considered: reference, moderate, and extreme.

C.3.3.1 Reference

The reference alternative corresponds to the situation already considered in Section C.3.1. That is, no compaction is performed other than that possibly provided by driving heavy equipment over the backfill and cap.

C.3.3.2 Moderate

A number of variations are considered for this alternative, depending upon the disposal alternative. For trenches and slit trenches, the backfill is assumed to be compacted using heavy construction machinery. For auger holes, a technique used at the Nevada Test Site (NTS) is assumed. Other procedures are considered for disposal methods involving concrete structures.

Soil compaction is a standard construction technique. Relationships can be developed for particular types of soil which relates the moisture content of the soil to the amount of compaction (the dry density of the soil). These relationships can be determined and graphed using laboratory techniques. For a particular soil, an optimum moisture content can be determined which results in maximum compaction (greatest dry density). In standard construction practice, specifications for compaction require the soil to be compacted near the optimum moisture content and to a dry density specified as a percent of the standard determined in the laboratory -- e.g., 90% of the standard (ASTM 1557) laboratory maximum density (Ref. 1).

A variety of equipment types may be potentially used depending upon the type of soil. Some of these may include sheepsfoot rollers, smoothwheel rollers, or vibrating baseplate compactors. Soil to be compacted would be applied in 6- to 12-in lifts and several passes made to compact each lift to the desired density. The depth of compaction available using such equipment is on the order of six feet (Ref. 1).

It is difficult if not impossible to achieve equivalent compaction at a low-level waste disposal facility. For one thing, shielding considerations would dictate applying larger thicknesses of backfill prior to compaction. The largest problems are probably the difficulty of achieving significant compaction of the waste containers and the large thickness of the waste mass which must be compacted. But in any case the increased compaction that could be achieved would probably help reduce subsidence of disposal cell covers -- at least over the first 5 or 10 years.

Equipment such as vibratory compactors, sheepsfoot rollers, or tractors would be of little use for compacting auger holes. For auger holes, then, an alternative compaction technique is assumed based upon a technique used for several years at the NTS to compact deep bore holes excavated as part of nuclear weapons test programs (Ref. 11). In this technique, a mixture of soil and granular material is loaded into a large steel funnel called a "stemmer." The stemming mixture pours through the funnel and falls into the augered hole. The free fall sifts the mixture so that it compacts readily. The consistency of the stemming material can be varied by varying the proportions of the mixture components. The rate at which the mixture pours through the stemmer is controlled by a gate, but otherwise there are no moving parts.

Compaction is more difficult for concrete disposal methods. For the concrete slit trench, concrete trench, and caissons, hand tampers or vibrators are assumed to be used. Heavy machinery may be used for the repack disposal option.

C.3.3.3 Extreme

The previous section discussed use of standard contruction techniques using heavy machinery (vibratory compactors, sheepsfoot rollers, etc.) to compact backfill into disposal trenches followed by compaction of the disposal trench cap. This compaction is expected to help compress disposed wastes and reduce voids, thus reducing settlement and subsidence problems, infiltration of water, and potential migration of radionuclides. Maintenance requirements would also be reduced. The depth of compaction achieved by these standard construction techniques is only a few feet, however. Thus, shallow compaction would not be expected to completely eliminate potential subsidence as long as a significant amount of compressible waste is disposed in the disposal trench.

Additional construction techniques which could be considered as expensive means to achieve very deep compaction (e.g., down to the bottom of a disposal trench), include pile driving and dynamic consolidation. Neither of these techniques have been used at operating disposal facilities, although both have been considered for disposal facilities which are no longer accepting waste (e.g., Refs. 24, 26, 31).

Pile driving as a means to densify deep soil deposits--particularly loose cohesionless soils--has been practiced for several years. In this technique, wood piles would be driven in a close grid pattern through the disposal trench cap and into the disposed waste. Compaction would be achieved through

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displacement of soil/waste mixture by the piles as well as by vibrations generated through driving the pile. After driving, the piles could be potentially removed and the holes filled with low compressive material such as cement or backfill. The piles could then be reused in another location. A problem with this would be that the piles would become contaminated as a result of contact with the waste materials. This contamination could then be available for transfer to workers or equipment or become dispersed into the air, thus becoming an occupational as well as an offsite radiation hazard. Procedures would therefore have to be developed to control the contamination. Such procedures would not be terribly difficult to implement but would tend to slow down operations.

The removed piles would eventually have to be disposed as radioactive waste. As an alternative, the driven piles could be cut off at ground level and covered with a compacted cap. This would result in increased expenses, however (Ref. 1).

Dynamic consolidation (or dynamic compaction) is a relatively new (25 years) construction technique which, while not previously used at radioactive waste disposal facilities, has been used to reduce settlement problems at landfills. The technique was developed in Europe by Menard, although its use in the United States has become more common. In practice, a large (5-40 ton) weight is dropped from a significant height (e.g., 20-100 ft) several times over a limited area. For an area such as a disposal trench, an optimum weight and drop height would first be determined. Then, a crane would drop the weight a number of times at several locations in a pattern across the trench cover surface. Depressions left by the weight would be filled in and additional passes over the trench surface may be made as desired and depending upon site-specific conditions.

The impact of the dropped weight is believed to cause partial liquefaction of granular and nonsaturated soil, which allows the soil mass to settle into a denser state. For saturated cohesive soils, it has been hypothesized that the shock waves and high stresses caused by repeated high energy impacts result in gradual liquefaction and consolidation of the soil. The method is reported to be effective to depths of 15 m (50 ft) and can achieve surface settlements of 5 to 15% of the deposit thickness (Ref. 1).

Other than the expense, the principal drawback to this compaction technique is the potential for expulsion of contaminated soil and waste. Depending upon the characteristics of the soil, the weight employed, and the drop height, depressions having depths of up to several feet may be produced. For most efficient compaction, the dropped mass should be allowed to penetrate a considerable distance into the disposed cell, which could result in violent expulsion of contaminated material into the air. This could cause a contamination problem for personnel and equipment, not to mention an airborne hazard both onsite and offsite. One way to reduce the potential for airborne spread of contamination would be to restrict the mass of the weight and the dropping height. However, this could also diminish the effectiveness of the compaction technique in that the depth of compaction could be reduced. Some of the voids would be bridged rather than eliminated.

Even if the technique was not 100% efficient, however, it can still be a useful technique for reducing maintenance requirements for disposed unstable wastes. A small test program has been conducted at Oak Ridge National Laboratory (ORNL) in which a low-level waste disposal trench previously filled with compressible

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material was subjected to dynamic compaction. The weight was relatively light (a few tons) and was raised to a height of a few tens of meters. Repeated drops were performed at the same spot until the desired depression was achieved (about 20 drops for a few meters depression). This approach was relatively conservative but greatly reduced wear on the crane cables, and also resulted in no expulsion of contaminated material. It has been estimated that the procedure eliminated on the order of 75% of the voids within the trench (Ref. 26).

Another drawback is that the above techniques would be most suitable only for trench disposal methods. They would appear to be unsuitable for deep auger holes, and may or may not be suitable for shallow auger holes or slit trenches. They would also appear to be most suitable for disposal trenches containing trash and other waste in a structurally unstable form. It would not appear to be cost-effective to use this technique to compact a disposal trench filled with stable waste forms such as concrete or vinyl ester styrene, and definitely counter productive for waste packages such as high integrity containers. It would also not appear to be appropriate for unstable wastes disposed using a grout backfill.

In any case, an example is considered in which dynamic compaction is used to compact certain disposal cells filled with structurally unstable waste forms. This compaction technique is assumed to be only applicable to trench disposal alternatives and slit trenches. Based on information given above, the compaction process results in an assumed average depression of about 1 m over the surface area of the disposal cell. This depression is filled in with compacted soil obtained from an onsite borrow. The costs are expected to be similar to those for piles.

C.3.4 Backfill Alternatives

Three basic backfill alternatives are considered in this section: natural (local) soil, granular material, and grout.

C.3.4.1 Natural Soil

This is the same alternative assumed for all the alternative disposal cell concepts considered in Section C.3.1. The backfill consists of soil previously excavated when constructing the disposal cells.

C.3.4.2 Granular Material

In the previous section, the backfill was assumed to consist of local soil, which depending upon site characteristics may vary from a very sandy to a very clayey texture. In this section, the backfill is assumed to consist of a granular material (sand, gravel) obtained from offsite. Disposal operations are carried out as before, with the exception that the sand or gravel backfill is substituted. The granular backfill would be expected to readily sift down into the interstitial spaces between waste packages and therefore help reduce the presence of voids in a disposal cell. This has been the experience at some existing disposal facilities.

In addition, the contact time with disposed waste for water percolating down through trench backfill may be greater for a backfill composed of lower permeable soils than for a backfill composed of higher permeable soils. This is

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because the speed of the percolating water may be higher for materials with higher permeability than for materials with lower permeability. Use of a sand backfill, then, could possibly allow percolating water to quickly flow past disposed waste to the bottom of the trench, thus reducing the contact time and the potential for leaching.

C.3.4.3 Grout

Another potential backfill alternative is to fill void spaces between waste packages with a cement grout. The grout would help stabilize the disposed waste and reduce subsidence of the disposal cell contents and disposal cell covers, thus reducing rainwater infiltration into the disposed waste. The decomposition of the waste would proceed at a slower rate. The grout would also reduce impacts to a potential inadvertent intruder, since it would increase the difficulty of excavating into the disposed waste. The cement fill would provide greater radiation shielding than ordinary soil backfill, and reduce the potential for airborne dispersion should the waste be intruded into. It would also increase the likelihood that an intruder would recognize that something was out of the ordinary and investigate. In the case of the concrete disposal methods (concrete trench, above-ground bunker, concrete slit trench, and concrete caisson), use of the concrete grout would turn the disposal cells into solid blocks of concrete interspersed with waste packages.

Grouting has not been performed at operating disposal facilities but has been tested as a stabilization technique at Maxey Flats, a facility that is no longer accepting waste for disposal. In this test, a section of an existing low-level waste disposal trench was blocked off and divided into an experimental section and a control section. This was done since the effectiveness of the grout technique in filling voids is uncertain. Delivery pipes were driven into the experimental section and cement grout pumped into the void spaces. Stakes were then planted in the trench caps of both the experimental and control sections as markers for subsidence operations. To date, no subsidence has been observed in either the experimental or the control section (Ref. 26).

To implement this alternative, a filter fabric layer is assumed to be laid down on the disposal cell floor to prevent clogging of drains and sumps. The waste is then emplaced in layers. After each layer is completed, grout is laid on and between the waste packages. For large or deep disposal cells such as trenches, the concrete trench, or the deep auger, grout is assumed to be delivered via flexible pipes mounted on booms. The pipes would be directed to void spaces between the waste packages at perhaps six to eight separate locations. The grout is pumped through the pipes until the grout level reaches the top of the first waste layer. After the first waste layer is grouted, additional waste emplacement could proceed. Each layer of waste would be similarly grouted.

For disposal cells having small areal dimensions, the grout is assumed to be delivered from cement mixers directly into the disposal cells using chutes.

Given the expense of the grout, the grout is assumed to be emplaced only up to the tops of the waste containers for most disposal technologies. These include trench disposal, slit trench disposal, auger disposal, and the repackage alternative. The remaining space between the waste and the original grade, if any, is backfilled with compacted soil. For the remaining concrete alternatives (concrete trench, concrete slit trench, above-ground bunker, and caisson), the grout is assumed to completely fill the residual space in the disposal cells.

In order that waste disposal operations not be halted during grouting, it would be necessary to operate with two or more disposal cells open concurrently. The labor force would also have to be augmented. Additional supplies and equipment would include grouting equipment (pumps, pipes, cement mixers, etc.), and cement. An increase in the storage area would also be needed for warehousing the cement prior to use.

C.3.5 Alternative Operational Procedures

This section considers four options for handling and placing waste in disposal cells: waste emplacement, waste segregation, layering, and bulk waste disposal.

C.3.5.1 Waste Emplacement

This alternative is only applicable for trench disposal methods as discussed in Sections C.3.1.1 and C.3.1.2. These trench disposal methods are in general use at existing low-level waste disposal facilities.

At these facilities, waste emplacement has been generally accomplished by either random disposal (including dumping or rolling containers into the disposal trenches, and placement of heavier items in a random fashion), or by stacked placement of items in some orderly or interlocking fashion. Stacked emplacement has been used to either maximize trench space utilization or provide waste-shielded "pockets" in which higher activity containers may be placed. Past practices at commercial disposal facilities have ranged from completely random to entirely stacked disposal, with current techniques generally characterized by a mixture of random and stacked emplacement. Variations of stacked emplacement have been used, including individual placement of stacked boxes, large right cylinders, and individual smaller (200 liter) drums in specific spots. In cavities formed by these first-layer containers, higher-activity wastes have been placed. Lower activity wastes were then randomly stacked or rolled, depending on the mode of off-loading that is most efficient, on top of the first-layer containers. The stacking height is dependent on the types of containers received, the capabilities of the waste handling equipment, and the backfill required to maintain desirable radiation levels.

An advantage of stacked rather than random placement of waste containers is that it enhances stability of the disposed waste, resulting from a reduction in trench void space and an associated decrease in the potential for subsidence. This promotes the integrity of the trench cover and reduces the infiltration of rainwater, thus reducing maintenance requirements as well as the potential for groundwater migration. Stacked emplacement is also estimated to improve the disposal efficiency from about 50% to about 75% or higher, resulting in an approximate 50% increase in trench capacity. Additional attractive features of stacked emplacement include a reduction of stresses on the integrity of waste containers, more control over high-activity containers, and use of other waste (instead of backfill) for shielding. Where trench space is at a premium and a sufficient fraction of the incoming waste packages have uniform configurations for stacking, it may be to the operator's advantage to use this method.

There are also disadvantages to stacking of waste containers. Stacking is a more labor-intensive effort compared with random placement. This not only increases the labor cost per unit volume, but raises worker radiation exposure

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levels proportionately. Where segregation of high-activity waste is not performed, trench radiation levels may at times prohibit workers from assisting in the desired positioning of containers.

C.3.5.2 Waste Segregation

Waste segregation is separation of waste materials at the disposal facility in accordance with specific characteristics. This may be according to isotopic content, chemical content, activity, container size, shape, or structural stability. Some forms of waste segregation are already in use at both government and commercial disposal facilities. Segregation of structurally stable and unstable waste forms in separate disposal cells is one practice growing out of the requirements in 10 CFR Part 61. Existing packaging, handling, and disposal restrictions on waste packages containing special nuclear material are another example. In addition, differing site acceptance criteria have forced some segregation of waste among the commercial facilities still in operation.

Two examples of wastes which are prime candidates for segregation are wastes having high concentrations of organic chemicals and chelating agents and wastes such as compressible low-activity trash for which long-term stability cannot be assured. Segregation of such wastes from other wastes will result in overall improvements to the potential for groundwater migration. With unsegregated waste disposal, nuclide migration from a disposal cell would be based on limiting isotopes and worse case conditions. With segregation, the most innocuous wastes having limited activity and short half-lives could be disposed in trenches engineered to contain that waste over its hazardous lifetime with a high confidence level. More hazardous and longer half-life wastes could concurrently be disposed in more appropriately engineered (and more expensive) disposal cells.

This concept is generally implemented in 10 CFR 61 by the segregation requirements for unstable Class A waste and the requirement to identify the unstable wastes on shipment manifests. Chelating agents are also required to be identified on shipment manifests if in significant quantity. In addition, there is a requirement to identify the principal chemical form of the waste, which can be used to identify scintillation liquids, the most common waste containing organic chemicals other than chelating agents. There is no specific requirement in 10 CFR Part 61 at this time, however, to segregate organic chemicals in disposal, although this requirement is implied for waste containing significant quantities of chelating agents.

Another possible practice could consist of segregating waste based on the surface radiation hazard of the waste packages. For wastes having high surface radiation levels which could require use of extensive backfill if mixed with the main body of waste, it may be more cost effective to use specially placed caissons or slit trenches which afford improved shielding with little backfill volume required. Use of this segregation technique may also reduce radiation exposures received by facility personnel from high-activity waste by 20% or more. This is because less direct exposure time is incurred while transferring high-activity waste to a deep excavation with vertical sides than to an open area where it must be covered to reduce lateral exposure rates. Slit trenches are in fact being used today for disposal of high surface radiation waste at the disposal facility at Barnwell, South Carolina. Caissons and tile holes have also been used in the past at other disposal facilities.

C.3.5.3. Layering

Protection against inadvertent intrusion may be accomplished by layering of the waste according to the relative hazard of the waste. The concept of layering involves placement of wastes having a higher potential hazard along the bottom of a disposal cell with wastes having a lower potential hazard emplaced on top. Typically, higher potential hazard waste would include waste packages characterized by high surface radiation levels or wastes that could pose a significant airborne hazard if disturbed by excavation.

For illustrative purposes consider the reference disposal facility (Section C.2). In the reference facility trench, only the bottom 7 m out of the 8 m excavated is used for disposal of waste. In layered waste disposal, the bottom 4 m of the excavation is assumed to be reserved for disposal of higher potential hazard waste material. Any remaining space in the bottom 4 m is used for disposal of lower activity waste. The 3 m of space available above the bottom 4 m is also used for disposal of lower potential hazard waste material. Thus, the inadvertent intruder would have to dig through at least 1 m of backfill, 1 m of cover, and 3 m of lower hazard waste before encountering waste that could result in a significant potential exposure. Excavation work that uncovered boxes and drums of low activity waste would probably discourage further excavation long before the more hazardous material was reached. Layered waste disposal would also help to reduce personnel exposures during disposal operations by providing additional shielding for wastes having high gamma radiation levels.

The option of layered waste disposal would not appreciably alter facility design, operations or labor requirements. However, there would have to be an adequate mix of lower hazard to higher hazard waste on hand to allow for successful implementation of the option. Maintaining an input of waste at this ratio would probably require either careful scheduling of input from waste generators, and/or implementing greater storage capability at the site. For example, if higher activity waste were to be received at the site at a rate equivalent to that of the lower activity waste, a fraction of it would be buried as it was received, while the remainer would be stored for future disposal (when sufficient lower hazard waste became available). It would also be necessary to have the capability of transporting the waste from the site waste storage area. Therefore, operational changes at the disposal facility could involve temporary storage of waste, additional coordination of waste receipt and emplacement, and transport of stored waste from the storage area to the disposal trench. Significant cost differences are estimated to include hiring of some additional personnel. No additional land would be committed to waste disposal.

C.3.5.4 Bulk Waste Disposal

This alternative could be used to dispose of bulk quantities of low activity waste such as contaminated dirt or bulk calcium fluoride waste from uranium fuel fabrication operations. Another example could include slightly contaminated concrete rubble from decommissioning nuclear power plants.

This option involves trench disposal. However, the bulk waste is assumed to be delivered in an unpackaged form in large dump trucks or equivalent. The waste is dumped into trenches up to a height approximately 1 m below the top of the trench. The trench is then backfilled and capped in the usual manner.

A major difference with previous waste disposal practices is that this disposal alternative provides very efficient use of available disposal space. Thus, the disposal efficiency for this alternative is assumed to be unity.

Theoretically, this disposal method should involve reduced costs, since the disposal operation is simplified. However, increased air monitoring would probably be needed. In addition, a disposal facility operator could possibly take the position that this disposal alternative involves "special handling of waste," and thus impose a surcharge.

C.3.6 Postoperational Alternatives

Postoperational activities include closure, surveillance, and institutional control; the first two activities are carried out by the site operator, the third by the site owner. In this report, closure of disposal cells is assumed to take place essentially as part of operations, and so only minor demolition and decontamination activities are anticipated during the closure period. Reference postoperational costs are calculated as a function of:

- disposal technology (i.e., whether or not the waste is disposed in a structurally stable form);
- disposal site environment (e.g., either arid or humid); and
- disposal site size.

These reference postoperational costs do not consider possible contingency expenses such as restabilization or leachate treatment. Such expenses may be considered, however, at the option of the code user.

C.4 DISPOSAL COSTS

This chapter presents the assumptions and rationale for determining base costs associated with siting, licensing, and operating a low-level waste disposal facility. Costs are given in 1984 dollars and are presented individually for the preoperational and operational periods. Postoperational costs are presented in Chapter C.5. The time value of these costs is presented in Chapter C.6.

C.4.1 Introduction

This section includes a description of the disposal and operational alternatives considered in more detail, as well as a description of the overall approach to performing the cost analysis.

Alternatives Considered Further

The preceding chapter outlined a large number of basic alternative disposal technologies, plus variations on these disposal technologies, that could be considered for waste disposal. It is believed that the outlined alternatives are fairly representative of the range of near-surface disposal variations that could be envisioned. In addition, several operational alternatives were addressed such as alternative technologies for disposal unit stabilization and capping, waste emplacement, and segregation. These only scratch the surface of the spectrum of possible disposal alternatives and variations thereof.

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Time and budgetary considerations dictate that only a limited number of disposal technology and operational alternatives can be fully considered in this report. Consideration of these alternatives furthermore implies two separate tasks: (1) developing numerical relationships to relate costs to factors such as waste volumes and activities, site environments, and so forth, (2) working the numerical relationships into a computer algorithm. Both tasks can become quite complicated.

Based on a number of considerations, cost relationships are developed for limited number of disposal technologies. Of these disposal technologies and variations, only 14 are programed into ECONOMY. These alternative disposal technologies are listed below, where the 14 technologies considered in ECONOMY are underlined:

-- large trench, humid environment (30m x 180m x 8m)
-- small trench, humid environment (10m x 60m x 6m)
-- large deep trench, arid environment (30m x 180m x 14m)
-- small trench, arid environment (10m x 60m x 8m)
-- small trench, arid environment, with slit trench extension (6m deep) for high activity waste
-- unlined auger, humid environment (3m dia. x 6m depth)
-- unlined auger, arid environment (3m dia. x 30m depth)
-- slit trench, humid environment (6m depth x 60m long)
-- slit trench, arid environment (6m depth x 60m long)
-- concrete trench, humid environment
-- concrete trench, arid environment
-- concrete slit trench, humid environment
-- concrete slit trench, arid environment
-- lined caisson, humid environment
-- lined caisson, arid environment
-- repackaged disposal operation, humid environment
-- repackaged disposal operation, arid environment

The operational alternatives that will be considered include the following:

-- cover alternatives (base, improved)
-- compaction alternatives (none, moderate, extreme)
-- backfill alternatives (none, soil, sand, grout)
-- operation alternatives (emplacement, segregation, layering)

Some of the operational alternatives are obviously not appropriate for all of the disposal technology alternatives. The relative applicability of the operational and disposal technology alternatives are summarized in the matrix formed by Figure C.13.

Approach to Cost Analysis

The approach taken in this report to perform economic analyses of disposal alternatives is somewhat different than that taken in previous efforts (Refs. 1, 2). Although a present value analysis will again be used, the increased complexity of the analysis dictates a new approach. Two of the factors to be considered include the need to vary the costs as a function of waste volume and design mix. Previously (Ref. 2), the analysis was geared to a fairly large disposal site which disposed on the order of 50,000 m^3 of waste

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	Reference Cover	Imp. Cover	No Compaction	Mod Comp.	Extreme Comp.	Soil Backfill	Sand Backfill	Grout Backfill	Random Disposal	Stacked Disposal	Unstable Segregation	Chemical and Chelates Segregation	Layering
Ref. Trench, Humid	X	X	X	X	X	X	X	X	X	X	X	X	X
Small Trench, Humid	X	X	X	X	X	X	X	X	X	X	X	X	X
Large Trench, Arid	X	X	X	X	X	X	X	X	X	X	X	X	X
Small Trench, Arid	X	X	X	X	X	X	X	X	X	X	X	X	X
Trench Ext, Small, Arid	X	X	X	X	X	X	X	X	X	X	X	X	X
Unlined Auger, Humid	X	X	X	X		X	X	X	X	X	X	X	X
Unlined Auger, Arid	X	X	X	X		X	X	X	X	X	X	X	X
Slit Trench, Humid	X	X	X	X	X	X	X	X	X	X	X	X	X
Slit Trench, Arid	X	X	X	X	X	X	X	X		X	X	X	X
Concrete Trench, Humid	X	X	X	X		X	X	X		X	X	X	X
Concrete Trench, Arid	X	X	X	X		X	X	X		X	X	X	X
Con. Slit Trench, Humid	X	X	X	X		X	X	X		X	X	X	X
Con. Slit Trench, Arid	X	X	X	X		X	X	X		X	X	X	X
Lined Caisson, Humid	X	X	X	X		X	X	X		X	X	X	X
Lined Caisson, Arid	X	X	X	X		X	X	X		X	X	X	X
Repack, Humid	X	X	X	X		X	X	X		X	X	X	X
Repack, Arid	X	X	X	X		X	X	X		X	X	X	X

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Figure C.13 Matrix of Operational and Disposal Technologies Considered in Detail

per year. Although some adjustment was allowable for different site volumes, some of the capital and other costs, which were considered reasonably fixed, were believed to be excessive for very small sites (e.g., on the order of 5,000 m^3 of waste per year, or less). Thus, there was perceived to be a need to accommodate very small sites accurately, since some sites in the future may only accommodate wastes from a few states.

The largest consideration was the design mix. The impact analysis methodology is capable of considering up to six different disposal technology designs on a given disposal site, and calculating and presenting impacts separately. This is because separate consideration is given to unstable Class A disposal, stable Class A disposal, Class B disposal, Class C disposal, Class D disposal (D1), and Class D disposal (D2).* The difficulty is in accurately determining the effect of all the various disposal technologies on the economic implications of the entire site. This is not possible with the approach taken in reference 2.

The solution, which is practicable given the use of computer technology, is to separate the base disposal costs into two general groups: preoperational (including capital) and operational. Rather than a lump sum approach, however, each group of costs is broken down into separate cost components, and a calculational algorithm is structured for each cost component that describes how the cost component varies as a function of waste volume and disposal design mix. Preoperational cost components include land costs, licensing costs, building construction, office equipment, and so forth; while operational cost components include personnel salaries, disposal cell materials, consumables, environmental monitoring, etc. After the costs are determined for each cost component, the cost components are added to form total preoperational and operational costs. The essential refinement over the previous approach is that the total preoperational and operational costs are calculated internally within the computer program rather than externally.

Other differences include a more sophisticated treatment of the timing of waste delivery to the disposal site. In past approaches (Refs. 1, 2), a constant average volume of waste was assumed to be delivered to the disposal facility during operation. (For example, 50,000 m^3/yr over 20 years operation.) The refinement in the waste projections calculated in Appendix A of this report allows consideration of waste volume generation on an annual basis. This allows consideration in the economic analysis of the timing of waste volume delivery to the disposal site.

C.4.2 Preoperational Costs

This section considers preoperational costs and is divided into two subsections. The first (Section C.4.2.1) presents the operational cost components, a description of the factors included in the cost component, and a detailed analysis of the variation of each cost component with respect to waste volume and disposal design mix. The second subsection (Section C.4.2.2) addresses the timing of

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*Two "Class D" waste classes are included to enable consideration of potential impacts from two possible groups of high-activity waste streams: those routinely generated today (D1), and those that are less routine or may be generated in the future (D2). For this report, Class D1 waste streams are assumed to be Class D waste streams originating from Groups I through V in Table 2-6 (see main text of this volume), while Class D2 waste streams are assumed to consist of Class D waste streams originating from Groups VI and VII.

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the preoperational expenditures over the preoperational period. In this report, a fixed 5-year preoperational period is assumed. However, the timing over this 5-year period during which capital and other expenditures occur has an influence in the present value analysis.

C.4.2.1 Preoperational Cost Components

Preoperational cost components are listed in Table C-4 along with a brief descriptive summary of the variance of each cost component. Each cost component and its variance is discussed below in more detail.

Land

Land costs are generally given as the product of the total acreage purchased and the price per acre. The total acreage purchased is given as:

$$A_t = A_a + A_d + A_b + A_c \qquad \text{(C-1)}$$

where,

A_t = total area purchased (acres)

A_a = administration area (9.1 acres)

A_d = disposal area (acres)

A_b = buffer zone area (acres)

A_c = contingency area (acres)

In this report, the administration area is assumed to be a constant 9.1 acres (3.7 ha) for all alternatives considered in the report. The operational area includes the area actually used for waste disposal (contains the disposal cells) plus a buffer zone around the disposal area. The disposal area is calculated by summing the surface area associated with each of the six possible waste classes (A-unstable, A-stable, B, C, D1, and D2).

$$A_d = 2.471\text{E-}4 \sum_{j=1}^{6} V_j \,/(\text{EMP} \times \text{EFF}_j \times \text{SEFF}_j) \qquad \text{(C-2)}$$

where

V_j = total volume of waste (m^3) disposed using the disposal technology associated with each waste class.

EMP = waste emplacement efficiency (dimensionless), which is the volume of waste emplaced in a disposal cell divided by the available disposal space in the disposal cell.

EFF_j = volumetric disposal efficiency (m), which is the volume of available disposal space in a disposal cell divided by the surface area of the disposal cell.

Table C-4. Preoperational Period Cost Components

Cost Component	Variation
Land	Vary with total surface area
Licensing · site screening and selection · site characterization · other studies · application preparation · environmental report preparation · preparation of procedures, manuals, etc. · NRC licensing fees · other permits (EPA, state, etc.) · legal fees · environmental monitoring · public outreach	Mostly constant
Administration · state administration · company administration · compact administration	Mostly constant
Land development	
· land preparation	Vary with total surface area
· access roads	Constant
· onsite roads	Vary with total surface area
· fencing	Vary with total surface area
· lighting	Vary with total surface area
Buildings	Vary with volume and design mix
Utilities installation · telephone · electricity · sanitary	Mostly constant (some variation with volume and design mix)
Health physics, office, and other light equipment	Vary with no. of personnel
Heavy equipment	Vary with volume and design mix
Startup overhead · salaries · relocation and travel · training	Vary with personnel no. and mix
Public outreach	Mostly constant
Engineering and design	% of selected components
Contingency	% of above components

$SEFF_j$ = surface utilization efficiency (dimensionless), which is the fraction of the total disposal area encompassing the disposal technology of concern which is occupied by the disposal cells (accounts for spacing between disposal cells).

2.471E-4 = conversion factor, m^2 to acres.

However, waste disposed using the slit trench extension is not considered in this calculation since it involves disposal of waste within a slit trench constructed at the bottom of a much larger trench. Also note that EMP must be constant for each waste class disposed together in the same disposal cells. (For example, if Class B and Class C wastes are disposed together in the same disposal cells, EMP must be identical for both waste classes).

The buffer area comprises the space between the disposal area and the boundary of the operational area. To calculate, the disposal area is assumed to be in the shape of the golden rectangle of Greek antiquity. Given this, it can be shown (See Section C.7.2.2) that the length (a) and width (b) of the disposal area (A_d) are given by:

$$a = [(A_d/k)(1+x)/x]^{\frac{1}{2}} \qquad \text{(C-3)}$$

$$b = A_d/ak \qquad \text{(C-4)}$$

where A_d is in acres, a and b are in meters, k = 2.471E-4 acres/m^2, and x is approximately 1.618.

Given this, the area of the buffer zone around the disposal site (A_b, in acres) is:

$$A_b = k(a + 2c)\ (b + 2c) - A_d. \qquad \text{(C-5)}$$

where (c) is the width of the buffer zone in m. Using the IBUF index (see Table 2-11), the code user may specify a buffer zone of either 30.5 m (100 ft) or 305 m (1000 ft).

The contingency area is miscellaneous acreage assumed to be purchased by the site operator. This could be used for storage, small site expansion, or for other activities. It is calculated as a fraction (30%) of the total operational area, or:

$$A_c = .3(A_d + A_b). \qquad \text{(C-6)}$$

It is recognized that at an actual site the contingency area could be considerably larger--perhaps up to a few hundred acres. However, the purchase of considerable additional land is believed to be either a site-specific or a company policy question. This is difficult to factor into an analysis methodology which is concerned with comparison of technical alternatives.

It is difficult to assess a generic cost per acreage. DOE stated in one 1983 document on low-level waste disposal economics (Ref. 12) that land costs ranged at that time from \$112/acre in the southwest to \$2,400/acre in the northeast. This was based on a U.S. Department of Agriculture document published in August 1980. DOE then assumed a 1981 average land value of \$1,400/acre. An average land cost of \$1,400/acre was also assumed in reference 2.

Another consideration is the possibility that the search for a suitable site will drive up the price of land. That is, there will probably be only a limited number of sites within a region that is suitable for technical and other reasons. This makes the land for the sites valuable property. Since landowners are aware of this, it is likely that the price for the land will be higher than if it was to be used for nonnuclear applications.

An average land cost of $2,000/acre is assumed in this report.

Licensing

This cost component is comprised of a number of subcomponents which account for costs involved with licensing studies, report preparation, fees, legal expenses, preoperational environmental monitoring, and public outreach. These costs are assumed to be relatively independent of the mode of site operation -- i.e., either as an operation for profit or as a public service enterprise.

Except for the subcomponent "other studies," these costs are also assumed to be constant for all disposal technologies and are also assumed to be independent of the disposal volume. This may not be totally true, since there are conceivable instances where licensing costs could be increased or decreased. For example, factors which could increase costs could include the possible need for considerable additional geohyrological studies caused by specific site conditions. On the other hand, it is possible that such costs could be reduced for a very small site, a site in which the waste is restricted to very low activity material, or for a site using a highly engineered design.

Such factors are difficult to consider in a generic analysis, although there is at least one example in the literature in which licensing costs were considered as a function of facility size (Ref. 12). In this DOE document, total licensing costs, which included site selection, characterization, application, manuals, legal fees, etc., were estimated to vary as follows:

Capacity (ft^3/yr)	Type	Cost ($ x 1000)
1,200,000	multiple waste type (a)	2,600
500,000	multiple waste type (a)	2,400
100,000	multiple waste type (a)	2,000
100,000	restricted waste type (b)	1,600
10,000	restricted waste type (b)	1,300
1,000	restricted waste type (b)	1,200

(a) Accepts all types and classes of low-level waste.

(b) Accepts only low activity, short half-life waste.

Studies, Reports, and Fees. For this report, these costs components are assumed to be as follows:

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Subcomponents	Assumed Costs ($ x 1000)
Site screening & selection	600
Site characterization	750
Other studies	*
Application preparation	500
Environmental report preparation	500
Preparation of procedures & manuals	100
NRC licensing fees	325
Other permits and fees (EPA, state, etc.)	250
Total:	3,025

*Cost for other studies are discussed below.

These costs were estimated based on information from similar types of economic analyses (Refs. 2, 12, 13). It is noted that these estimated costs are higher than similar costs estimated by most of the others (Refs. 2, 12, 13, 14), but are within the range estimated by an operator of an existing disposal site (Ref. 15). These (1982) costs estimated for site screening and characterization, and license applications and fees, totaled $3,500,000 for a 1.2 million ft^3/yr site (Ref. 15).

The subcomponent "other studies" includes seismic and other analyses. Such analyses are assumed to be relatively minor for non-engineered disposal technologies (trenches, slit trenches, unlined augers). Such analyses are assumed to be more detailed for underground engineered disposal technologies (concrete trench, lined caisson, repackaged disposal). This is because the engineered structure is relied upon to provide structural stability. The "other studies" subcomponent is therefore assumed to vary as follows:

Disposal Technology	Cost ($ x 1000)
non-engineered	200
underground engineered	400

Legal Fees. These are estimated as an average of $200,000 per year over the 5-year preoperational period, or a total legal expense of $1,000,000. This is twice as high as the total legal expense estimated in one reference (Ref. 13) and consistent with an estimate given by the operators of an existing low-level waste disposal facility (Ref. 15).

Environmental Monitoring. The preoperational environmental monitoring program establishes a baseline level of background radioactivity, including ranges in concentrations of specific isotopes. These costs are in addition to those associated with site characterization (e.g., geologic profiles), and are projected to be expended during the last 4 years of the preoperational period.

Costs for installation of monitoring equipment are assumed to be generally included under the licensing cost component. Wells used for site characterization (geologic profiles, water tables profiles), for example, can be readily used for monitoring wells. The principal contributors to these costs are therefore assumed to be personnel costs and sampling analysis costs. Annual costs for a radiation safety technician are projected to be as follows:

	25K
30% benefit	+ 7.5K
	32.5K
50% overhead	+16.25K
	$48.75K

Assuming the annual dedication of 1/4 man-year, plus approximate sample costs of $12,500 (50 samples per year at $250/sample), this calculates to about $25,000/year over 4 years. These costs are assumed to be constant for all disposal alternatives.

Public Outreach. It is recognized that radioactive waste disposal is a subject area that is of great concern to many people, and yet the principles behind safe disposal activities are understood by relatively few. Thus, a relatively significant effort is believed to be called for by disposal site operators and/or state administrators to contact and inform the public of the proposed scope of operations. Outreach activities can include public meetings, pamphlets, and radio and television addresses. This cost component is believed to be relatively independent of the size of the facility or scope of operations. It could perhaps be reduced for some specific cases such as a very low volume disposal site, a highly engineered disposal site, or a site accepting only low levels of contamination. This is a bit speculative, however, and so a constant annual expense of $100,000 is budgeted throughout the preoperational period.

Administration

This cost component reflects expenses of an administrative nature spent during the preoperational period. It consists of personnel costs while managing preparation of license applications, performing environmental surveys, and so forth. It consists of two basic subcomponents: company administration costs and state or compact administration costs.

In this report, the site is assumed to be operated for a profit as a business. The site operator would perform site selection activities and environmental studies, and prepare and submit a disposal facility application to the NRC (or appropriate state regulatory agency) for review. The costs for regulatory review of the license application have already been considered as part of licensing fees. However, other state or compact administration costs must be considered.

In the past, potential operators of disposal facilities went through some manner of site selection process and submitted applications largely on their own initiative. Other than costs associated with state regulatory agencies, state costs were probably principally related to management of the lease for the disposal facility land (generally state owned). Given the compacting process, however, a state or compact could solicit operators to prepare a preliminary application which would include such information as company qualifications and proposed designs. Actual site selection could follow selection of an operator. In this case, costs would result from selection of the operator through the above competitive process. The costs for the compact administration must also be considered. These compact administration costs may vary depending upon a number of highly state- or compact-specific factors, and may potentially be quite large.

Corporate Administration. This consists of two sets of costs: corporate support costs and local administration. Support is provided by the main corporate office, while local administration costs are those associated with an office located in the capital city of the state in which the disposal facility is sited. Local office costs include office rental and equipment, personnel salaries and benefits, travel, relocation, and other expenses. Also included are expenses associated with an onsite trailer which is used by the company during much of the preoperational period (e.g., for sampling, survey, other activities). Many of the activities carried out in the local office are moved to the site itself during operations, although the office is retained at a reduced level.

Corporate support costs are assumed to total $100,000 per year for each year of the preoperational period. About half of the local administration costs are associated with personnel costs. These are calculated assuming a team of six persons which is assumed to be involved essentially full time during the preoperational period. These include a project manager, two senior engineers (or project managers), two junior engineers (or project managers), and a clerical assistant. In addition to the base salaries, a 30% benefit is assumed. This benefit rate represents indirect labor costs such as pensions, health insurance, and other benefits. (Alternative ways to calculate benefit costs are also possible.) The breakdown of these costs is as follows:

Personnel			Costs ($ x 1000)
1 Senior project manager	@ 55k		55.0
2 Senior engineers	@ 35k		70.0
2 Junior engineers	@ 24k		48.0
1 Clerical	@ 12k		12.0
			$185.0
		30% benefit	+ 55.5
			$240.5

Except for the first preoperational year, other overhead expenses (e.g., office rent and supplies, travel) are assumed to total approximately $100,000 per year. An additional cost of $30,000 is assumed during the first year of preoperation to cover relocation expenses for personnel in the local office.

Compact Administration. This cost component will depend upon compact-specific conditions and is thus very difficult to estimate. A compact administration cost of $200,000/yr is somewhat arbitrarily assumed.

Land Development

This cost component accounts for initial site construction other than utilities installation and buildings. Subcomponents include:

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- initial land preparation
- access roads
- onsite roads
- fencing
- lighting

Initial land preparation. This consists of clearing sufficient land to construct buildings, roads, and disposal cells. There are two ways of approaching this. One is to initially clear sufficient land to begin operations (e.g., 40 acres), and then to continuously clear additional land as part of operations. This reduces preoperational costs by spreading some of it over the operational period. The other approach would be to perform most or all of the initial land preparation as part of the preoperational period. This is prior to receipt of any radioactive waste, and so there would be no need to take special contamination control precautions. Such operations should therefore be easier to carry out. The latter approach is taken here, and costs are taken to be proportional to land area as follows:

$$\text{Costs} = C_u(A_a + A_d + A_b) \qquad \text{(C-7)}$$

where, A_a, A_d, and A_b have been defined previously, and

C_u = unit cost ($/acre) for clearing a moderate amount of scrub.

Based on information from reference 16, unit costs are assumed to be $1450/acre for clearing and grubbing medium undergrowth. This unit cost includes labor and equipment costs, and also costs for hauling to a dump.

Access roads. The costs for access roads are entirely site specific and are furthermore independent of site design and waste volume. Thus, a fixed cost is assumed for all alternatives. This is taken to be an improved gravel road about 10 m wide. Costs include those for cut and fill, grading, drainage control, surface placement, and any necessary bridging. The total road length depends on the proximity of existing and usable roadways. The total cost is assumed to be $100,000.

Onsite roads and parking areas. Like access roads, onsite roads and parking areas are assumed to be of the improved gravel variation. This is the practice at existing disposal facilities. To calculate, a perimeter road is assumed to be constructed along the inside of the fence for security and other reasons. Two other roads bisecting the facility lengthwise are also assumed to be constructed.

The perimeter and other roads are therefore a function of land area. Parking areas in the administrative and operational areas are assumed to vary depending on the number of personnel.

Perimeter and onsite roads are assumed to be about 10 m wide, and are calculated as a cost per unit length. The installed cost per unit length is assumed to be about $50/m for cut and fill, grading, and a layer of gravel (Refs. 16, 17). The total length is a function of the size of the operational area, or:

$$\text{Length} = 4(a + 2c) + 2(b + 2c) \qquad \text{(C-8)}$$

where $a = [(A_d/k)(1+x)/x]^{1/2}$, $b = A_d/ak$, A_d = disposal area (acres), $k = 2.471E-4$ acres/m², $x \approx 1.618$, and c = width of buffer zone (m). All these parameters have been discussed earlier in this section.

Parking areas are assumed to cost about \$5/m², including cut and fill, grading, and a layer of gravel. The total parking area is calculated based on the assumption that the required area is essentially governed by the size of the operation, which in turn governs the number of personnel employed. Parking areas are required for site personnel, waste delivery and other vehicles, maneuvering room around facility buildings both in the administrative and operational areas, and so forth. Roughly 40 m² of space is assumed per person in the administrative area, including parking access, and maneuvering space. This is doubled to account for required parking and maneuvering space in the operational area, resulting in a total "effective" area per person of 80 m².

Fencing. Fencing costs are calculated as the product of the operational area perimeter length times an average price per length. For an average height of 8 ft, the cost per length of chain link fencing is about \$30 per linear foot (Ref. 18), or \$9.14/m. The perimeter length is assumed to be a function of land areas as follows:

$$\text{Length (meters)} = 2a + 2b + 8c \qquad \text{(C-9)}$$

where all parameters are discussed above.

Lighting. Lighting costs are calculated in a similar manner as fencing costs--i.e., a function of the above perimeter length. Lighting fixtures are assumed to be installed at intervals of about 30 m. Installed costs per lighting pole, assuming an aluminum pole 25 ft high with a 6 ft arm, are assumed to be about \$1700 per pole (Ref. 18).

Buildings

Building costs are difficult to project. At least six buildings are assumed for all alternatives, although the size and complexity of the buildings are estimated to vary significantly depending upon waste volume and disposal facility design mix. These six buildings are based on those assumed for the reference disposal facility in the final Part 61 EIS (Ref. 2). In this document the six buildings are sized for a 50,000 m³/yr waste throughput, and a total site payroll of 80 persons:

Buildings	Reference Area (m²)	Reference EIS Costs* (1980 \$/m²)
Administration	750	376.6
Health Physics/Security	800	484.4
Warehouse	470	269.1
Garage	420	269.1
Waste Activities	560	538
Storage Shed	80	108

*As constructed.

A seventh building--the waste processing and repack building--is added for the large scale waste repackaging disposal option, provided that waste disposed by this method exceeds 100 m^3/yr. Finally, extensive implementation of some activities such as concrete disposal technologies or waste grouting will require annual production of large quantities of cement. A separate facility is assumed as discussed below.

Administration. The administration building contains office space for site management and other administrative and support personnel working at the site. Therefore, the administration building is assumed to vary depending upon the maximum number of administrative and support personnel employed over the life of the site. Based on a reference 750 m^2 for 32 administrative and support personnel, the building size is assumed to vary by 12.5 m^2 for every administration and support person either added or subtracted. After the building size is determined, the costs are determined by multiplying the size by the unit cost. This is assumed to be \$376.6/m^2 as inflated from 1980 to 1984 dollars using the ratio of the producer price indices (PPI) for capital equipment. This turns out to be 1.23, resulting in a cost of \$463/m^2 (Ref. 19).

Health Physics/Security. This building houses a security section, a small counting room, health physics offices, a change room/locker room, a lunch area, and a supply room. It is assumed to vary in size depending upon the maximum number of workers and guards employed over the operating life of the site, where guards are assumed to vary totally based on waste volumes and workers are assumed to vary depending upon both waste volume and disposal design mix. In any case, a reference building size of 800 m^2 is assumed for 48 guards and workers, and 12.5 m^2 is added or subtracted for each guard or worker different from the base 48. However, a minimum size of 300 m^2 is also assumed. Unit costs are assumed to be \$484.4/m^2 as inflated using the PPI ratio, or 596/m^2.

Warehouse. The warehouse is used to store supplies used on the site. One would expect the size of the warehouse to vary depending upon the total number of site personnel as well as the complexity of site operations. However, the two factors are complimentary (a more complex operation implies more personnel) and so the variance is assumed to be limited to the maximum number of site personnel employed over the life of the site. Based on a reference size of 470 m^2 for a reference 80 people, 6 m^2 is assumed to be added or subtracted for each person varying from the base 80. However, a minimum size of 200 m^2 is also assumed. Unit costs are assumed to be \$269.1/m^2 as inflated using the PPI ratio, or \$331/m^2.

Garage. The garage is required for maintenance of vehicles on an as-needed (rather than continuous) basis, and the size is therefore assumed to be a stair step function of the total number of onsite vehicles and equipment. A minimum size of 210 m^2 is assumed, and the variance is projected as follows:

Equipment Number	Garage Size (m^2)
0-13	210
14-39	420
40+	630

Unit costs are assumed to be \$331/m^2.

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Waste Activities. This building houses several functional areas, including: (1) a large item decontamination bay, (2) a control room for the decontamination bay, (3) a liquid treatment system, (4) a waste solidification, packaging, and overpackaging area, (5) a supply room, and (6) a small waste storage area. This facility is assumed to be of constant size (560 m^2 or \$371,000) for all disposal technologies and waste volumes. This is because, like the garage, it is used on an as-needed rather than a continuous basis.

Storage Shed. This is a small (80 m^2), portable building used to store supplies and miscellaneous tools within the operational area. It is assumed to be of constant size for all alternatives. The building cost is thus assumed to be \$10,600.

Waste Processing and Repack Building (WPRB). This building was discussed in Section C.3.1.12, and is assumed for the repackaged disposal option only if the total volume waste disposed by this method exceeds 100 m^3/year. The central features of the WPRB include separate waste repackaging operations for low and high surface radiation level waste containers (waste that can and cannot be contact handled), a nondestructive analysis station for waste content verification, a 1000-ton compactor, and temporary storage areas for both incoming and repackaged waste.

The WPRB cost is expected to be a slight function of waste volume. The building cost is given as the following equation:

$$\text{Cost} = A_1 + V_w A_2 \qquad \text{(C-10)}$$

where V_w is the total annual volume of waste (m^3/yr input) processed through the WPRB, A_1 is the base cost independent of volume considerations, and A_2 is a building cost per unit waste volume. For this report, A_1 is assumed to be \$600,000, while A_2 is assumed to be \$20/m^3.

Cement Plant. For some alternatives, bulk cement production will be required to be used as either concrete or cement grout. Based on expected stringent regulatory requirements for quality assurance of the concrete or grout product, it is anticipated that this material will need to be produced on site rather than be obtained from an outside source. Therefore a dedicated cement plant is assumed when either an alternative using cement is assumed, or grout is used as a backfill. Based on information from a number of sources, the cement plant is assumed to cost \$160,000 as installed (Refs. 20-23).

Utilities Installation

This cost component covers installation of telephone equipment, a potable water supply, and electrical lines and switches. Site water is assumed to be provided by an onsite well. These costs are expected to be only a small function of facility size and design mix. This is because much of the installation costs would be relatively independent of size, and may depend more upon the facility location and access to utility connections. An example may be installation of electrical lines and connections. The bulk of the costs relate to installation of the power transmission lines, which are totally site-specific, rather than the onsite distribution network (which would be more a function of facility size and design).

In an [illegible]e, assumed installation costs (includes material and labor) for water, [illegible]ephone, and electricity are listed below (Ref. 22):

Component	Cost($)
200 ft. well with casing	3,200
· well	3,200
· pump	1,500
· 1000 ft of 2" pipe	1,700
· 1000 ft of trenching	4,000
· 60,000 gal tank	80,000
· booster pump	2,000
	(92,400)
3,000 ft elec. power lines	
· transformer	1,100
· poles	27,000
· cables	35,000
· misc. controls, panels	11,000
	(64,200)
3,000 ft. telephone line, coaxial	($10,000)
	$166,600

Some small additional costs would be associated with installation of a package sanitary system (septic tank plus piping). Based upon the size of the site and the disposal technologies considered, site personnel could range from about 20 to about 100 persons. Within this range, sanitary system costs would not be expected to significantly vary.

A total constant cost of $200,000 is therefore assumed for the Utilities Installation cost component.

Health Physics, Office, and Other Light Equipment

This cost component covers purchase of various types of light equipment used in the various buildings by disposal site personnel. This equipment includes administrative building typewriters, files, and furnishings; health physics/security building furnishings; portable fire extinguishers; and so forth. It also covers survey meters and other miscellaneous radiation detection equipment such as 2-π and 4-π counters and radiation alarms.

This cost component is expected to be a variable of the number of site personnel. Based on input from other sources (Refs. 13, 25), this cost component is estimated as roughly $3,500 per site employee. A 10-year equipment replacement schedule is also assumed.

Heavy Equipment

This cost component is one of the largest contributors to preoperational costs, and is also one of the most difficult to calculate. In the draft and final EIS for 10 CFR Part 61 (Refs. 1, 2), the uncertainties connected with pricing heavy equipment and determining replacement schedules were noted, and so all heavy equipment was simply assumed to be leased. This arrangement was recognized to be not totally satisfactory, however. A more traditional approach would be to take equipment costs as a capital expense, and to also assume an orderly replacement schedule over the operational life time. Additional difficulties then arise from attempting to assess equipment needs for any waste volume and any design mix. Some disposal technologies require specialized equipment.

One possible approach for estimating the principal pieces of heavy equipment is to first estimate for each disposal technology considered, the total number of heavy equipment-days required per unit waste volume for each type of equipment. All activities would be included, including disposal cell construction, waste emplacement, backfilling, capping, and maintenance. The total equipment-days would be then divided by the total working days available during a year. (250 working days per year seems reasonable.) A certain amount of dead or maintenance time would be also assumed. (80% availability appears to be reasonable.) This results in the total number of each type of heavy equipment--i.e., for each type of heavy equipment (e.g., a crane):

$$\text{Number} = K \sum_{i=1}^{I} \sum_{n=1}^{6} V_n E_{ni} \qquad \text{(C-11)}$$

where:

V_n = waste volume (m^3) for disposal technology considered;
E_{ni} = equipment-days per disposal technology (n), operational activity (i), and waste volume (m^3); and
K = use factor.
= (1/250)(1/0.8) = 5.0E-3

The above approach is complicated but is extremely flexible and can be readily accomplished using computer technology. A problem is that it is not possible to precisely identify on paper all of the activities that will be carried out at a given site. Some pieces of equipment may be needed only occasionally at indeterminable frequencies. An example may be a backhoe or an electric welding unit. A related problem is that the calculations for each piece of equipment will be expressed in terms of decimals rather than integers For example, it may be determined using the above equation that 2.03 pan scrapers of a given size are needed during an operational year. This leaves the question of whether it is better to purchase 3 pan scrapers, to purchase 2 pan scrapers plus rent a third for a fraction of the year, or to purchase 2 pan scrapers of slightly larger size. This question can be readily resolved for a specific disposal facility and waste volume, but is very difficult on a generic basis.

Another consideration is that some types of equipment are more dependent upon the total number of specific personnel types. These include 4-wheel drive vehicles, pickups, and other company vehicles. The total number of 4-wheel drive vehicles would potentially be based on the total number of guards. The total number of pickup trucks would be based on the total number of site workers. The total number of miscellaneous company vehicles would be based on the total number of administrative personnel.

For the purposes of this report, a combined approach is used. Some pieces of equipment are always assumed to be needed for all waste volumes and disposal technologies, and these are listed in Table C-5. At least one of each of the equipment pieces listed in Table C-5 is assumed to be required. Additional

Table C-5. Required Equipment

Equipment	Cost ($1000)	Replacement schedule (yrs)	Calculational method
Bulldozer	200	10	Eqn C-9
Front end loader	100	10	Eqn C-9
Dump truck	40	10	Eqn C-9
Pan scraper	60	10	Eqn C-9
Motor grader	100	10	Eqn C-9
Backhoe	100	10	Eqn C-9
40-ton crane	150	10	Eqn C-9
100-ton crane	500	10	Eqn C-9
Forklift, small	30	10	Eqn C-9
Water truck (a)	55	10	Eqn C-9
Pickup truck (b)	10	5	(c)
4WD truck (b)	15	5	(d)
Sedan	12	5	(e)
Accessories (g)	100	10	1 only
Air monitors	(f)		(f)

(a) With spraying attachments.
(b) With communication and safety equipment.
(c) Based on total number of site personnel as discussed in text.
(d) Based on total number of guards as discussed in text.
(e) Based on total number of administrative personnel as discussed in text.
(f) At least three are assumed to be installed for each disposal option. Costs for these are already included. Additional units are installed at a cost of $1,000 apiece as discussed in the text.
(g) This includes various small pieces of equipment such as slings and other rigging, a dragline bucket, an electric welding unit, and so forth.

pieces of equipment are assumed according to the type of calculation listed in the table. Table C-6, on the other hand, lists "optional" equipment pieces -- i.e., equipment that is assumed to be purchased based on the particular disposal technology. For most equipment types, the total number is based on use of equation C-9. Others are calculated as discussed below.

Pickup trucks, 4WD vehicles, and corporate vehicles (sedans) are based on particular types of site personnel. For example, 4WD vehicles would vary depending upon the number of guards employed, while corporate vehicles would vary depending upon the number of administration personnel. Ultimately these can be related to annual waste volume received. The assumed relationship is listed below.

Annual Waste Volume (m^3/yr)

Equipment	2,000	5,000	20,000	35,000	50,000
Pickup truck	2	2	3	4	5
4WD vehicle	2	2	3	4	5
Sedan	1	1	1	2	2

Table C-6. Optional Equipment

Equipment	Cost ($1000)	Replacement schedule (yrs)	Calculational method
Auger rig	400	10	Eqn C-9
Stemming unit	4	10	Eqn C-9
Paving machine	90	10	Eqn C-9
Tandem roller	35	10	Eqn C-9
Compactor	80	10	Eqn C-9
Forklift, large	40	10	Eqn C-9
Farm tractor	20	10	1 only (a)
Hand tamper	1	5	Eqn C-9
Cement truck	45	10	Eqn C-9
1 CY cement bucket	5		1 only (b)
Cem. pump & pipes	100		Eqn C-9
Yard tractor	120	10	(c)
Flatbed trailer	20	10	(c)

(a) Assumed to be needed if the disposal facility is located in a humid environment.

(b) Assumed to be needed for only the concrete trench and concrete slit trench options.

(c) Based on waste volume with variances for the repack option.

The flatbed trailers are used to ease problems with waste scheduling and delivery. Frequently, waste can be unloaded onto a trailer from a delivery vehicle and temporarily stored. The delivery vehicle can then be checked out of the site, easing congestion. For all waste disposal options except the repack option, the following relation is assumed:

Annual Waste Volume (m^3/yr)

Equipment	2,000	5,000	20,000	35,000	50,0000
Yard tractor	0	0	1	1	1
Flatbed trailers	0	0	1	2	3

For the repack option, all repackaged waste is assumed to be delivered to the disposal cells using trailers. The relationship is given below.

Annual Waste Volume (m^3/yr)

Equipment	2,000	5,000	20,000	35,000	50,000
Yard tractor	1	1	1	2	2
Flatbed trailers	1	1	2	3	5

Costs per equipment type were assumed based on a number of sources, including costs as estimated by engineering guide manuals as well as costs estimated by other authors for similar projects (Refs. 12, 13, 16-18, 27, 28). Frequently, wide variations were observed in costs for similar types of equipment.

Startup Overhead

This cost component covers startup overhead costs incurred during the last preoperational year. It is composed of a number of subcomponents including personnel salaries, relocation and travel expenses, and training. This cost component is assumed to be a variable dependent upon the number of personnel employed during the first year of site operation. Based on review of other cost estimates (Refs. 13, 15), it is approximated as about 50% of the personnel salary and benefit costs during the first year of operation.

Engineering and Design

This cost component is assumed to be 10% of the total costs for land development, buildings and utilities installation.

Contingency

This cost component is calculated as 20% of all of the above cost components.

C.4.2.2 Timing of Cost Components

Since a present value analysis is used to perform the economic calculations, the results will be significantly influenced by the timing of costs during the preoperational period. The preoperational period can be roughly divided into a prelicensing period (site screening and selection, site characterization, and license application preparation), a licensing review period, and a site construction period as illustrated on Figure C-14 (after Ref. 15). Based on this figure and on other considerations, Table C-7 was prepared.

Table C-7 provides for each preoperational cost component an estimate of the percentage expenditure over the 5-year period. In this table, the total costs and annual percentages of expenditure refer to base 1984 costs prior to consideration of inflation.

Most costs associated with licensing are projected to occur in the second year while almost all construction costs are projected to occur in the fifth year. Some minor construction (e.g., fences, unimproved access roads, some utilities) is projected to occur prior to site licensing. Such construction, however, is related to requirements for preoperational site characterization and monitoring, as opposed to the extent of construction that would be necessary for receipt and disposal of waste. Some cost components are more or less fixed, and such costs are totaled and listed in the table. Other cost components are variable depending upon personnel, disposal volume, or design considerations, and these are noted appropriately in the table.

It should again be noted that the above cost estimates and algorithms are approximations at best, and that an analysis for a real site should consider regional and site-specific data.

C.4.3 Operational Costs

The assumed operational cost components are listed in Table C-8, along with a summary description of the variation. A more detailed discussion of each cost component is included below.

Activity	Year 1	Year 2	Year 3	Year 4	Year 5
Site Screening and Selection	XXXXXX				
Site Characterization	XXXX	XXXXXXXXXX			
Preparation of License Application		XXXXX			
Licensing Review			XXXXXXXXXX	XXXXXXXXXX	
Site Construction					XXXXXXXXXXX

Figure C.14 Approximate Siting, Licensing, and Construction Timeframes

Table C-7. Cost Expenditure (Percent) per Preoperational Year

Cost Component	Total Cost ($ x 1000)	Preoperational Years 1	2	3	4	5
Land	*	20	2	2	2	74
Licensing	*					
· site screening & selection	600	100				
· site characterization	750	33	67			
· other studies	*		33	33	34	
· application preparation	500		100			
· environmental report preparation	500		100			
· prep. of procedures & manuals	100				75	25
· NRC licensing fees	325				100	
· other permits (EPA, state, etc.)	250				100	
· legal fees	1,000	20	20	20	20	20
· environmental monitoring	100		25	25	25	25
· public outreach	500	20	20	20	20	20
Administration						
· company	2,232.5	20	20	20	20	20
· state and/or compact	1,000	20	20	20	20	20
Land development	*		10		10	80
· land preparation						
· access roads						
· on-site roads						
· fencing						
· lighting						
Buildings	*					100
Utilities installation	200					100
Health physics and other light equipment	*					100
Heavy equipment	*					100
Startup overhead	*					100
· salaries						
· relocation and travel						
· training						
Engineering and design	#					
Contingency	##					

*Costs variable.

#Calculated as 10% of annual costs for land development, buildings, and utilities.

##Calculated as 20% of total annual costs listed above.

Table C-8. Operational Period Cost Components

Cost Component	Variation
Salaries	Vary with volume and design mix
Administration · state · company	Mostly constant
Regulatory costs	Mostly constant
Consulting and studies	Mostly constant
Legal fees	Mostly constant
Public outreach	Mostly constant
Personnel training	Vary with personnel number
Disposal cell materials	Vary with volume and design mix
Environmental monitoring	Vary somewhat, mostly with volume
Personnel monitoring	Vary with personnel number
Equipment replacement · heavy · light	Vary with volume and design mix
Miscellaneous expenses	Vary with waste volume
Utilities	Vary with volume and design mix
Heavy equipment operating expenses	Function of equipment type and use
Maintenance	% of selected components
Insurance	Mostly constant
QA and compliance testing	Vary with volume and design mix
Contingency	% of above components

Salaries

Personnel costs are one of the largest contributors to the total operational costs, and are naturally a function of the number and variety of site personnel, which are in turn assumed to vary as a function of waste volume and disposal technology design mix. In this report, site personnel are assumed to be grouped into three basic groups: administration, support, and workers. In general, the number and type of administrative and support personnel are assumed to be a function of waste volume, although some consideration is given to disposal technology. Use of more sophisticated (engineered) disposal technologies implies the need for additional administrative and support personnel to comply with additional quality assurance requirements.

Worker type and number are assumed to be a function of the disposal technology used, and amount of waste disposed by each disposal technology. The basic approach is to determine for each disposal technology the number of worker-hours required for excavation, construction, disposal, capping, and maintainance, as a function of waste volume. The total is then summed over all disposal technologies utilized. Total personnel requirements are then determined by dividing by a realistic number of working days per year (220 out of 365), and including an efficiency factor to account for breaks and other downtimes (360 minutes out of 480 minutes).

The assumed distributions of administration and support personnel are listed in Table C-9 as a function of waste volume. Using the assumed individual salaries as also listed in Table C-9, the following annual costs are assumed for administration and support:

Staff Costs ($ x 1000)	Annual Waste Volume-m^3/yr (ft^3/yr)				
	(70,620) 2,000	(176,550) 5,000	(706,200) 20,000	(1,275,850) 35,000	(1,765,500) 50,000
Administrative	168	168	203	387	426
Support*	102	133	236	282	354
Subtotal*	270	301	439	669	780
30% Benefit	81	90.3	131.7	200.7	234
Total	351	391.3	570.7	869.7	1,014

*Does not include costs for QA technicians and radiation safety technicians.

For annual waste volumes less than 2,000 m^3/yr, a constant cost of $265,000/yr is assumed. For other waste volumes, costs (and personnel number) are interpolated between costs for the appropriate waste volume column. Total administration costs are determined by adding benefit costs which are calculated as 30% of base costs.

The third group of site personnel, workers, varies depending upon waste volume and design mix, and are assumed to include radiation safety technicians and quality assurance technicians. Requirements for such personnel are very difficult to estimate, given the large variation in disposal technologies considered in this study, and for which little U.S. experience is available. The approach taken is to first identify the principal factors contributing to personnel

Table C-9. Administration and Support Personnel Distribution as Function of Waste Volume

Staff	Individual Salary ($x1000)	Annual Waste Volume-m^3/yr (ft^3/yr)				
		(70,620) 2,000	(176,550) 5,000	(706,200) 20,000	(1,235,850) 35,000	(1,765,500) 50,000
Administration:						
*Site manager	50	1	1	1	1	1
*Asst. site manager	40	0	0	0	1	1
Executive secretary	15	0	0	0	1	1
*Radiation safety officer	35	1	1	1	1	1
*Foreman	28	1	1	1	1	1
*Operations manager	38	0	0	0	0	1
*QA & safety supervisor	30	1	1	1	1	1
Office manager	38	0	0	0	1	1
*Security chief	25	1	1	1	1	1
Records chief	15	0	0	0	0	1
Customer service coordinator	24	0	0	0	1	1
Contracts coordinator	24	0	0	0	0	1
*Personnel manager	32	0	0	0	1	1
*Regulatory affairs manager	35	0	0	1	1	1
*Site engineer	35	0	0	0	1	1
Subtotal		5	5	6	12	15
Support:						
*Guards	18	3	4	5	6	6
Waste shipment schedulers	20	1	1	2	2	3
Billing/Accounting	15	1	1	2	3	4
Secretarial	13	1	2	4	5	6
*Junior engineers	24	0	0	1	1	2
Subtotal		6	8	14	17	21
Total		11	13	20	29	36

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*Personnel contributing to annual personnel monitoring costs.

estimates, and then to estimate individual personnel man-day requirements for each factor based on waste volume and disposal technology. Then, the total individual man-days are summed and divided by the total number of annual working days. This is assumed to be 220 days per year at a 360/480 = 75% efficiency factor. The number of individuals per job category is thus summed, with any fraction thereof rounded upward. This sum is multiplied by the assumed annual salary per job category, and then 30% benefit calculations are made in the usual way.

The principal factors contributing to personnel estimates are taken to be as follows:

- Vehicle check in and out
- Disposal cell construction
- Cement plant operation
- Waste processing
- Waste placement and backfilling
- Covering
- Facility and grounds maintenance and radiation surveys
- Vehicle and equipment maintenance
- Environmental monitoring
- Quality assurance

Each of these factors is discussed below and assessed as a function of personnel man-days per unit waste volume and disposal technology. Given holidays and probable requirements for an on-site inspector, 250 annual working days are assumed. The workers considered in the ensuing formulations, plus assumed base annual salaries, are given below:

Worker	Base Annual Salary ($ x 1000)
Heavy equipment operator	25
Unskilled laborer	15
Skilled laborer	25
Surveyor	50
Quality assurance technician	25
Radiation safety technician	25

Total annual base salaries thus determined are then added to those for administrative and support personnel as given earlier based on waste volume considerations. These base costs are finally increased assuming a 30% benefit.

Vehicle Check In and Out. This factor can vary considerably depending on a number of considerations. These include the volume of waste received, the types of packages the waste is delivered in, the surface radiation levels of the waste packages, and the types of vehicles the waste is delivered in. The approach taken is first to estimate for each waste stream the distribution of waste volume among a limited number of waste container sizes and types. From this, estimates are made of the distribution of waste containers among a limited number of transport vehicles. Finally, the time required to process a single shipment into and out of the disposal site is determined; this time is then distributed among members of a team. Further complicating this assessment is the fact that the containers which are used, as well as the shipment vehicles, will change depending upon the surface radiation level of the waste package.

In this report, five general waste packages were considered as well as six general waste transport vehicles and overpacks. These packages, transport vehicles, and overpacks are listed below:

Waste Containers	Vehicles and Overpacks
Large wooden boxes - 128 ft^3	Vans (V)
Small wooden boxes - 16 ft^3	Flatbed trailers (FB)
55-gallon drums - 7.5 ft^3	Shielded trailers (ST)
Small liners - 50 ft^3	Large shielded casks (LC)
Large liners - 170 ft^3	Small shielded casks (SC)
	1-drum shielded casks (1D)

Some vehicles are capable of transporting more than one type of waste container, while other vehicles are of limited application. The assumed maximum numbers of waste containers per vehicle and overpack are listed in Table C-10.

Table C-10. Assumed Maximum Number of Waste Containers per Vehicle and Overpack

Vehicle and overpack	Container	No. in Shipment	Shipment Volume ft^3 (m^3)
Van	Large box	3	384 (10.9)
	Small box	36	576 (16.3)
	Drum	70	525 (14.9)
	Small liner	11	550 (15.6)
Flatbed trailer	Large box	4	512 (14.5)
Shielded trailer	Large box	3	384 (10.0)
	Small box	36	576 (16.3)
	Drum	70	525 (14.9)
Large shielded cask	Large Liner	1	150 (4.2)
	Small box	6	96 (2.7)
	Drum	14	105 (3.0)
Small shielded cask	Small liner	2	100 (2.9)
	Drum	6	96 (2.7)
1-drum cask	Drum	1	7.5 (0.2)

The use of a particular combination of waste container, vehicle, and overpack for a given waste stream depends upon: (1) an assessment of what types of waste containers are typically used for that waste stream, depending also upon the waste processing performed prior to shipment, and (2) the surface radiation levels of the waste containers when shipped. The assumed distribution of each waste stream over each container type is listed in Appendix B as a function of waste spectrum. As part of each particular volume distribution for each waste stream, a very approximate assessment is made of the surface radiation levels of each container type. (See Chapter 3.0 of the main text.) This assessment is used to categorize each container into one of three handling "care levels." These care levels have been dubbed regular care, special care, and extreme care.

The care level is used to restrict some of the options used for waste packaging and, more importantly, determines the use of particular combinations of particular vehicles and overpacks. The assumed waste containers available for each care level, plus the distribution of vehicles per package type, is given below:

Table C-11. Container Care Levels and Vehicle Distributions

Container and Care Level	Vehicle and Overpack (a)	Percent of Given Container Shipped by Given Mode
Regular Care:		
Large Box	Van	24
	FB	76
Small Box	Van	100
Drum	Van	100
Small Liner	Van	100
Large Liner	LC	100
Special Care		
Large Box	ST	100
Small Box	ST	96
	LC	4
Drum	ST	48
	LC	51
	SC	1
Small Liner	SC	100
Large Liner	LC	100
Extreme Care		
Drum	SC	51
	1D	49
Small Liner	SC	100
Large Liner	LC	100

(a)FB = flatbed trailer; ST = shielded trailer; LC = large shielded cask;
SC = small shielded cask; 1D = 1 drum shielded cask.

Thus, the numbers of different types of containers, vehicles, and overpacks are determined for each waste stream and care level. These are summed over all the waste streams delivered to the site. Once the total number of waste vehicles annually arriving at the site is determined, the personnel requirements are estimated by multiplying by the man-day requirements per shipment.

Man-day requirements per shipment are estimated as follows:

Personnel	Man-Days/Shipment
Radiation safety technicians	.15
Quality assurance technicians	.05
Skilled laborer	.05
Total	.25

Disposal Cell Construction. This personnel factor considers a number of unit operations going into construction of disposal cells. For non-engineered disposal methods such as trench, slit trench, or auger hole disposal, unit operations include gross excavation, finish (hand) work, excavation and placement of drains and sumps (when applicable), placement of standpipes and surface access boxes, and placement of markers and monuments. For disposal methods involving concrete, typical additional operations include laying substrates, placement and removal of formwork, setting rebar, pouring concrete, concrete finishing, backfilling and compacting around completed disposal cells, and placement of an asphaltic runoff layer (where applicable). These operations do not include waste emplacement, backfilling, and capping.

Unit construction personnel man-days and machine-days are listed as a function of disposal technology in Table C-12. Construction requirements were estimated using standard engineering cost guides (Refs. 16-18). QA technicians are assumed to spot-check construction work, and so requirements are assumed to vary depending upon the extent of the construction project. A general assumption is made that 1 QA technician man-day is required per 100 man-days (total) for heavy equipment operators, skilled laborers, and laborers.

Cement Plant Operation. Personnel requirements are based on an assumed grout or cement production of about 50 m^3/day. Man- and machine-day requirements per 50 m^3 of cement or grout are as follows:

Personnel/Machine	Requirement
HE Operator	1
Skilled Laborer	2
Laborer	1
QA Technician	0.04
Front End Loader	1

Waste Processing. This factor considers personnel requirements for operation of the waste process and repack building (WPRB) as part of the large-scale waste repackaging disposal alternative. Base personnel requirements and costs are estimated as a function of input waste volume as follows.

Table C-12. Unit (per Disposal Cell) Construction Machine and Man-Days per Disposal Technology

Disposal Tech. Basis Max Waste Vol.(m^3) Emplacement	Ref. Trench, Humid per trench 16,490/24,370 R/S	Small Trench, Humid per trench 1090/1640 R/S	Large Trench, Arid per trench 30,100/45,140 R/S	Small Trench, Arid per trench 1480/2210 R/S	Trench Ext. Arid per ext. 215 S	Auger, Humid per 40 grp. 997 S	Auger, Arid per 40 grp. 6,150 S	Slit Trench, Humid per trench 739 S	Slit Trench, Arid per trench 752 S
Personnel:									
·HE Operator	122.44	9.89	191.14	10.51	0.85	23.78	78.62	8.66	8.54
·S. Laborer	100.33	9.50	147.02	8.23	0.58	30.66	39.87	7.77	5.69
·Laborer	123.27	10.20	191.14	10.62	0.85	26.00	79.55	9.05	8.62
·Surveyor	0.50	0.50	0.50	0.50	0.25	1.00	1.00	0.25	0.25
·QA Technician	3.5	0.3	5.3	2.9	0.02	0.8	2.0	0.25	0.2
Equipment:									
·Bulldozer	32.98	2.57	29.85	2.66		2.39	2.39	0.29	0.29
·Loader	24.56	1.98	37.94	2.10	0.15	3.44	10.28	1.57	1.45
·Pan scraper	51.67	3.61	51.67	4.37					
·Dump truck	97.22	7.50	147.02	8.23	0.58	10.66	39.87	5.77	5.69
·Motor grader	12.13	1.21	4.67	0.89		5.95	5.38	0.38	0.38
·Backhoe	0.59	0.21			0.70			6.42	6.42
·Crane alone	0.50	0.30	68.11	0.50					
dragline bucket	0.50	0.30	68.11	0.50					
·Cement truck									
·Paver									
·Tandem roller									
·Compactor									
·Auger rig						12.00	60.00		

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Table C-12. (continued)

Basis Emplacement Max Waste Vol. (m^3)	Con. Slit, Humid per 6 grp. $ 937	Con. Slit, Arid per 6 grp. $ 937	Con. Trench, Humid per trench $ 4900	Con. Trench, Arid per trench $ 4900	Caissons, Humid per 40 grp. $ 51.2	Caissons, Arid per 40 grp. $ 51.2	Repack, Humid per trench $ 12,500	Repack, Arid per trench $ 12,500
Personnel:								
·HE Operator	135.92	134.51	137.89	136.46	60.67	60.05	130.34	174.50
·S. Laborer	501.44	495.44	675.52	673.52	154.93	154.93	240.91	278.77
·Laborer	181.00	176.08	241.96	236.96	71.85	67.51	247.02	291.46
·Surveyor	1.00	1.00	0.50	0.50	0.50	0.50	0.50	0.50
·QA Technician	8.2	8.1	10.6	10.5	2.9	2.8	6.2	7.5
Equipment:								
·Bulldozer	29.85	29.85	26.12	26.12	16.24	16.24	40.85	52.22
·Loader	11.92	11.92	11.95	11.95	6.30	6.30	21.35	30.81
·Pan scraper	14.46	14.46	16.44	16.44	7.11	7.11	33.37	55.53
·Dump truck	47.52	47.52	47.59	47.59	25.09	25.09	85.02	122.88
·Motor grader	4.61	4.61	6.22	6.22	3.12	3.12	22.65	23.07
·Backhoe								
·Crane	49.84	49.84	60.64	60.64	13.33	13.33	0.24	0.99
alone	46.30	46.30	55.93	55.93	12.71	12.71		
dragline	0.18	0.18	0.23	0.23	0.18	0.18		
bucket	3.36	3.36	4.48	4.48	0.44	0.44	0.24	0.99
·Cement truck	4.96	4.96	12.76	12.76	0.61	0.61	10.58	10.58
·Paver	0.70		0.71		0.62			
·Tandem roller	0.70		0.71		0.62			
·Compactor	23.83	23.83	15.02	15.02	13.58	13.58	11.88	11.88
·Auger rig								

Occupation	Individual Salary ($ x 1000)	Annual Waste Volume (m^3/yr x 1000) <10	10<20	20<30	30<40	40<50	≥50
WPRB Foreman	28	1	1	1	1	1	1
S. Laborer	25	1	1	1	2	2	2
Laborer	15	1	1	2	2	2	2
QA Technician	25	2	2	3	3	4	5
Rad. Safe. Tech.	25	2	3	3	4	5	6
Total pers.:		7	8	10	12	14	16
Total Salary ($ x 1000):		168	193	233	283	333	383

Base salaries are then increased to include a 30% benefit. All of these employees are badged.

Waste Emplacement and Backfilling. This factor considers personnel requirements for placement of waste in disposal cells followed by backfilling where appropriate. It does not include placement of a final cover (cap), although for concrete disposal technologies it does include replacement of the concrete lids. This factor is influenced by a number of considerations, including:

- Waste packaging
- Waste surface radiation levels
- Waste transport vehicles and overpacks
- Disposal technology
- Method of waste emplacement (random or stacked)
- Method of backfilling (none, native soil, sand/gravel, or grout)
- Backfill compaction (none, moderate, or extreme)
- Other operational procedures (layering of high activity waste, segregation of unstable waste or waste containing organic chemicals or chelating agents, or bulk waste disposal)

Emplacement. The general approach used to estimate personnel and machine requirements is based on an extension of the above approach for estimating requirements for vehicle check in and out. As discussed above, each waste stream is assigned a packaging mode distribution and a care level based on physical and radiological characteristics and past history. This in turn influences the numbers and types of shipment vehicle used for that waste stream. However, once the total number and types of individual waste packages are determined for each stream, then emplacement requirements are determined by multiplying each package by the man-minutes required to emplace a given waste package in a disposal cell. These man-minutes are then distributed among individual members of an emplacement team. Summing the individual man-minutes over all waste streams received at the site results in total individual man-minute requirements. These man-minute requirements are converted to man-day requirements by dividing by 400 working minutes per individual per day. This is based on an assumed 8-hour day and an assumed 50 minutes of realistic work per hour. A similar calculation is performed for machine requirements.

The basic unit man-minute package emplacement requirements are based on an assessment of former operations at the Maxey Flats disposal site and are thus principally applicable to shallow land burial (Ref. 29). Separate sets of unit man-minute values are tabulated for random and stacked disposal. Emplacement time in this study includes waste receipt, vehicle inspections and preparation of the delivery vehicle for unloading (opening the cask closures, etc.).

The basic emplacement crew in this study is assumed to be as follows (Ref. 29):

Personnel	Number
Radiation safety technician	2
Yard supervisor	1
Heavy equipment operator	2
Skilled laborers	2
General laborers	3
Total:	10

The study assumes familiarity with site equipment by all crew members. The following basic equipment mix is also assumed in the Maxey Flats study (Ref. 29):

Equipment	Number
Yard tractor	1
Flat bed trailers	2
Bulldozer	1
Pan scraper	1
Front end loader	1
20-ton crane	1
100-ton crane	1
Fork truck, small	2
Fork truck, large	1
Yard FWD vehicles	2
Total:	13

The above assessment includes equipment and manpower that would be needed for backfilling the emplaced waste, which in reality would take place at the same time as waste emplacement. These two activities will be artificially separated here, however, for calculational convenience.

The above assessment is directly applicable to the reference disposal trench, the small humid trench, and the small arid trench. For the large arid trench and all other disposal technologies, some modifications must be made. A significant difference is that for all remaining disposal technologies except the repack option, the technology design precludes extensive use of fork lifts, which were assumed to be used for placement of much of the regular care waste--especially boxes and stacked drums. Essentially all waste (especially regular care waste) must then be emplaced using cranes. For special and extreme care waste, surface radiation levels would normally require increased use of cranes for emplacement in any case. Except for the large arid trench, all remaining disposal technologies assume use of stacked disposal.

Based on the above considerations the reference man-minutes, emplacement crews, and equipment assumed for emplacement are listed in Tables C-13 and C-14. These are applicable to all disposal methods except the repack option and vary principally based on whether fork lifts are suitable for emplacement of a portion of the waste. Large boxes and waste arriving in casks are assumed to be most efficiently emplaced using cranes in any place. For drums and small boxes arriving in trailers or vans, additional emplacement time is assumed to be required when use of fork lifts is restricted.

For the waste repack option, all waste is repackaged into only two types of containers. The containers are of similar dimensions, provide shielding, and may be stacked in place using cranes or fork lifts. In this case, unit emplacement requirements are assumed to be 50 man-minutes per container. These unit emplacement requirements are assumed to be distributed among equipment and crew as follows:

Fractional Distribution

Equipment	Crew Personnel
0.25 - 40 ton crane	0.2 - Radiation safety technician
0.25 - 100 ton crane	0.1 - QA technician
0.50 - forklifts, large	0.2 - Heavy equipment operators
	0.2 - Skilled laborers
	0.3 - Laborers

For segregated waste, the above man-minutes are assumed to be multiplied by a factor of 1.1 per unit volume of waste. This accounts for the need to operate more than one disposal cell at once, and the time lost while shifting personnel and equipment. For layered waste disposal, the man-minute requirements of the Q.A. technician are assumed to be multiplied by 1.1. This accounts for extra work required to ensure proper placement.

Backfilling. Unit man- and machine-days for backfilling operations are a function of the particular disposal technology used and of the following:

- Waste emplacement (random, stacked)
- Backfill material (soil, sand/gravel, grout)
- Compaction (none, moderate, extreme)

The option of either random or stacked waste emplacement is assumed to be only applicable to the four trench disposal alternatives. For all other disposal technologies, only stacked emplacement is considered.

The three backfill options are applicable to all disposal technologies. For concrete disposal methods, however, use of soil as a backfill is probably not a wise choice, although the alternative is retained in any case. For the grout option, the grout is only taken up to approximately the top of the waste. This is done to save costs when used for disposal methods such as trench or slit trench. The remaining space to original ground level, if any, is filled with compacted earth. For concrete options using lids, the grout is emplaced up to the bottom of the lids.

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Table C-13. Man- and Machine-Minutes for Emplacement per Container

		Man and Machine Minutes for Disposal Per Container			
		Ref. Trench, Small Humid Trench, and Small Arid Trench		All Other Disposal Methods Except Repack Option	
Care Level and Container	Overpack(a)	Random	Stacked	Random	Stacked
Regular Care:					
Large Box	Van	100	120	100	120
	FB	37	60	37	60
Small Box	Van	8	12	9	13
Drum	Van	3	12	4	13
Small Liner	Van	68	83	68	83
Large Liner	LC	600	720	600	720
Special Care:					
Large Box	ST	150	180	150	180
Small Box	ST	13	20	15	22
	LC	125	150	125	150
Drum	ST	5	12	7	14
	LC	43	88	43	88
	SC	100	156	100	156
Small Liner	SC	300	360	300	360
Large Liner	LC	600	720	600	720
Extreme Care:					
Drum	SC	100	156	100	156
	1D	300	360	300	360
Small Liner	SC	300	360	300	360
Large Liner	LC	750	900	750	900

(a) FB = flatbed trailer; ST = shielded trailer; LC = large shielded cask; SC = small shielded cask; 1D = 1-drum shielded cask.

Table C-14. Equipment and Crew Assumptions for Waste Emplacement

Parameter	Fractional Distribution Ref. Trench, Small Humid Trench, and Small Arid Trench	 All Other Disposal Methods Except Repack Option
Crew Personnel:		
Rad. safety tech	0.2	0.2
Q.A. tech	0.1	0.1
Heavy equip. op.	0.2	0.2
Skilled laborer	0.2	0.2
Laborer	0.3	0.3
Equipment Mix:		
40-ton crane	0.2	0.67
100-ton crane	0.2	0.33
Fork truck, small	0.4	
Fork truck, large	0.2	

The essential difference between the first two compaction options (none and moderate) is that in the second case, machinery is used to tamp and vibrate the backfill in order to help work the backfill material down into the interstitial voids between the waste packages. The third compaction option involves use of dynamic consolidation, a technique in which a 40-ton weight is dropped repeatedly on the ground to be compacted, and is only assumed for trench disposal, slit trench disposal, the slit trench extension, and the shallow augered hole. Use of dynamic consolidation is also assumed to be incompatible with grouted waste disposal.

Unit man- and machine-days are calculated as a function of disposal technology and the operational alternatives discusssed above, and are based on input from a number of sources (Refs. 16-18, 28, 30, 31). For most disposal alternatives (trench, slit trench, trench extension, repack, humid auger) backfill material is assumed to be stored on site in a temporary storage pile, loaded using front end loaders into trucks, driven to the disposal cell in use, dumped, and spread using bulldozers. Compaction, when called for by user option, is accomplished using machinery such as a large vibratory or wobbly wheeled compactor. Machine capacities are assumed to be as follows:

Machine	Capacities (CY/day)
Bulldozer	600
Loader	1200
Truck	300
Compactor	500

The same capacities are assumed to be appropriate for soil as for sand or gravel. This is done for simplicity although one would expect a different daily capacity for sand or gravel.

Grout is assumed to be mixed at an onsite plant and brought to the appropriate disposal cell using mobile mixers. For shallow augers, the slit trench, and the trench extension, the grout could probably be delivered directly using a mobile mixer and a chute. The larger disposal cell dimensions characteristic of trench disposal implies use of pumps and delivery pipes. The pipes would be flexible and mounted on booms. Using cement mixers alone, grout delivery is assumed to be at a rate of 150 CY/day, and to require one skilled laborer, two general laborers, and one radiation safety technician. For large disposal cells, a cement pumping and delivery unit is added along with one heavy equipment operator and one laborer.

Soil or sand/backfill for the concrete slit trench, concrete trench, or caisson is assumed to be delivered to the disposal cell in the usual way. A loader (100 CY/day) is assumed to be used (rather inefficiently) to help dump and spread backfill, as are general laborers (10 CY/day/laborer). Radiation safety technician requirements are assumed to be the same as those for heavy equipment operators. If compaction is desired, this is carried out using hand tampers (24 CY/day/laborer). Grout is delivered by the alternative methods discussed above.

For deep augers, backfill may be either dumped into the shafts using bulldozers or, by user option, emplaced using a stemming rig. In this case, soil or other

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backfill is dumped using front end loaders (800 CY/day) into a hopper which granulates the material into fine particles. The fine particles falling into the shaft fill voids and provide compaction. Stemming units are assumed to be mounted on vehicles or trailers to allow rapid set up and break-down, and are assumed to require two laborers to operate at an average rate of 100 CY/day. Requirements for radiation safety technicians are assumed to be the same as for heavy equipment operators. If grout is used, it is emplaced using pumps and flexible pipes up to the top of the disposed waste. For the standard deep auger, the remaining meter of space is backfilled using a bulldozer and a compactor. For the auger with the 10-m plug, the remaining space is filled with stemming material.

For dynamic compaction, the operation requires use of a 60-80 ton crane and a 40 ton weight. A crew consisting of one heavy equipment operator, two radiation safety technicians, and three general laborers is assumed. Operations are assumed to be carried out at a rate of about 325 m^2 per machine-day.

In general, operation of bulldozers, loaders, large compactors, cranes, and concrete delivery units is assumed to be by heavy equipment operators. Trucks are assumed to be driven by skilled laborers. Man-day requirements for general laborers are taken as equal to skilled laborers, unless laborer requirements are augmented based on particular operational variations.

These unit man- and machine-day requirements are generalized to account for the possibility of a varying amount of waste per disposal cell. That is, due to regulatory restrictions on some types of waste, it may be necessary to make less than maximum use of a disposal cell. To model this, it is assumed that the sizes of the constructed disposal cells do not change, but that the height of the emplaced waste mass within the disposal cell may change. This leads to a variable quantity of backfill within a disposal cell depending upon the amount of waste placed in the disposal cell.

For calculational purposes, it is useful to separate the backfill emplaced in the interstitial spaces between the waste packages ("waste backfill"), from the backfill emplaced between the disposed waste and the top of the disposal cell ("plug backfill"). This is illustrated in Figure C.15 for a trench disposal cell. The waste backfill volume is given by:

$$WB = \frac{(1-EMP) * WD}{EMP} \qquad \text{(C-12)}$$

where

WB = waste backfill volume (m^3);
EMP = emplacement efficiency; and
WD = waste volume emplaced within disposal cell. (WD may be the total of a number of different waste classes.)

The plug backfill volume is given as the total empty volume of the disposal cell minus the volume of the waste and waste backfill. The volume of the waste plus waste backfill is given as WD/EMP. The total empty volumes (assuming a 1:4 side slope) for trenches, slit trenches, and the slit trench extension are given below along with surface dimensions (a,b), subsurface dimensions (c,d), the maximum usable disposal cell depth (h), and the disposal cell thickness (DTK):

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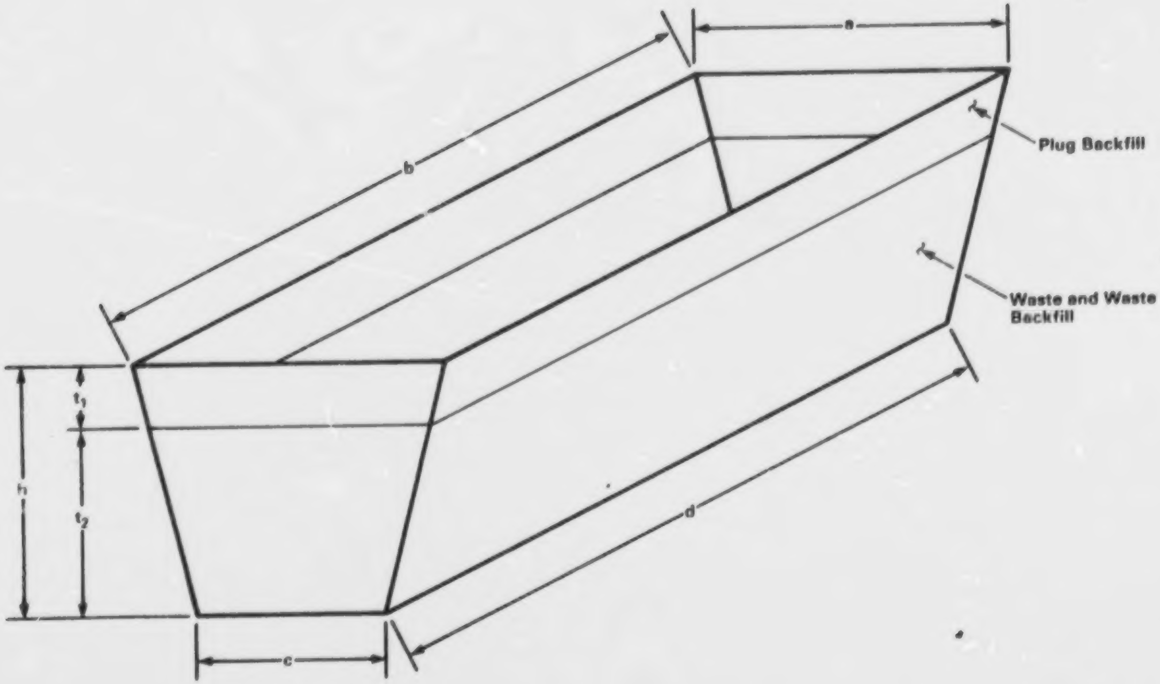
a
b
Plug Backfill
Waste and Waste Backfill
t_1
t_2
h
d
c

Figure C.15. Illustration of Waste and Plug Backfill for Trench Disposal Method

Disposal Technology	Empty Vol.(m^3)**	a*	b*	c*	d*	h*	DTK*
Ref. Trench, Humid	38,326	30	180	26.15	167.15	7.7	6.7
Small Trench, Humid	2,763	10	60	7.15	57.15	5.7	4.7
Large Trench, Arid	65,539	30	180	23	173	14	13
Small Trench, Arid	3,533	10	60	6	56	8	7
Trench Ext., Arid***	132	4	10	1	7	6	5
Slit Trench, Humid	1209	4	60	1.15	57.15	5.7	4.7
Slit Trench, Arid	1227	4	60	1	57	6	5

*Dimensions in m

**Considers volume taken up by ramp, if any, leading into the disposal cell.

***Three slit trench extensions can be excavated within each normal sized (small arid) trench, resulting in a total empty volume of 396 m^3.

For all other disposal technologies, the disposal cells have vertical side walls, where AD is the surface area (m^2) of a horizontal plane of waste within the disposal technology. Values for the disposal cell empty volume, AD, and h, the maximum usable disposal cell depth, are given below:

Disposal Technology	Empty Vol.(m^3)	AD (m^2)	h (m)	DTK(m)
Auger, Humid*	40.3	7.07	5.7	4.7
Auger, Arid*	212.1	7.07	30	29
Con. Slit, Humid**	219.4	37.5	5.85	5.7
Con. Slit, Arid**	219.4	37.5	5.85	5.7
Con. Trench, Humid	5,033	860.4	5.85	5.7
Con. Trench, Arid	5,033	860.4	5.85	5.7
Caissons, Humid***	1.46	0.28	5.2	5.2
Caissons, Arid***	1.46	0.28	5.2	5.2
Repack, Humid	28,690	4,250	6.75	6.75
Repack, Arid	28,690	4,250	6.75	6.75

*Values per single auger; for 40 auger group, AD = 282.74 m^2, empty volumes = 1612 m^3 and 8,484 m^3 respectively.

**Values per single trench; for 6 trench group, AD = 225 m^2, empty volume = 1,316 m^3.

***Values per single caisson; for 40 caisson group, AD = 11.31 m^2, empty volume = 58.2 m^3.

Once WB and PB are determined, man- and machine-day requirements may be determined. Assuming either soil or sand/gravel backfill with no compaction, individual man- and machine-day requirements may be determined by multiplying the quantity (WB + PB) by the appropriate information in Table C-15.

Assuming that moderate compaction is performed, then individual man and machine-day requirements are determined by multiplying (WB + PB) by the appropriate information in Table C-16.

For grout disposal, grout emplacement man- and machine-day requirements are calculated separately from those for any backfill placed above the waste. For

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Table C-15. Unit Man- and Machine-Day Requirements for Soil or Sand Backfill with Minimum Compaction

Man- or Machine-Days per m^3 Backfill

Man or Machine	All Except Con. Trench, Con. Slit Trench, and Caisson	Con. Trench, Con. Slit Trench, and Caisson
Personnel:		
· H.E. Operator	3.27E-3	1.42E-2
· S. Laborer	4.36E-3	4.36E-3
· Laborer	3.27E-3	1.45E-1
· QA technician	1.09E-4	1.64E-3
· Rad. Safety Technician	3.27E-3	1.42E-2
Equipment:		
· Bulldozer	2.18E-3	0
· Loader	1.09E-3	1.42E-2
· Dump truck	4.36E-3	4.36E-3

Table C-16. Unit Man- or Machine-Day Requirements for Soil or Sand Backfill with Moderate Compaction

Man- or Machine-Days per m^3 Backfill

Men or Machines	All except Arid Auger, Con. Trench, Con. Slit Trench, and Caisson	Arid Auger	Con. Trench, Con. Slit Trench and Caisson
Personnel:			
· H.E. Operator	5.89E-3	1.58E-2	1.42E-2
· S. Laborer	4.36E-3	4.36E-3	4.36E-3
· Laborer	5.89E-3	2.89E-2	6.87E-2
· QA Technician	1.61E-4	4.92E-4	8.73E-2
· Rad. Safety Technician	5.89E-3	1.58E-2	1.42E-2
Equipment:			
· Bulldozer	2.18E-3	0	0
· Loader	1.09E-3	2.73E-3	1.42E-2
· Dump truck	4.36E-3	4.36E-3	4.36E-3
· Compactor	2.62E-3	0	0
· Stemmer	0	1.31E-2	0
· Tamper	0	0	5.45E-2

humid augers, slit trenches, the trench extension, concrete slit trenches and caissons, unit man- and machine-day requirements per m^3 of grout are as follows (grout vol = WB m^3):

Personnel		Equipment	
S. Laborer:	8.72E-3	Cement Mixer:	8.72E-3
Laborer:	1.74E-2		
Rad. Safety Tech:	8.72E-3		
QA Technician:	2.61E-4		

For other disposal technologies, grout must be pumped into place using delivery pipes mounted on booms. Man- and machine-day requirements per m^3 of grout are as follows:

Personnel		Equipment	
H.E. Operator:	8.72E-3	Cement Mixer:	8.72E-3
S. Laborer:	8.72E-3	Cement Pump and Pipes:	8.72E-3
Laborer:	2.62E-2		
Rad. Safety Tech:	8.72E-3		
QA Tech:	4.36E-4		

Backfill above the grout is given by PB m^3. For most disposal technologies, the backfill is composed of soil or sand/gravel, in which case Tables C-15 and C-16 are used. For concrete slit trenches, caissons, and concrete trenches, the remaining volume is filled by grout, in which case the above two grout tables are used.

For dynamic compaction, the waste disposal cells are first backfilled according to Table C-15. Dynamic compaction operations, however, are a function only of disposal cell surface area; unit requirements are presented in Table C-17.

Covering. This factor considers personnel requirements for placement of final covers (caps) over disposal cells. Two different general disposal cell covers are considered: a base cover and an improved cover. Other differences include whether the disposal technology considered is appropriate for a humid or an arid environment. The base cover in general consists of naturally excavated soil which is indifferently compacted. The humid improved cover contains low permeability layers which are compacted in strips. For a humid environment, disposal cell covers are assumed to be planted with short rooted vegetation (plus top soil in the improved case), while for an arid environment, disposal cell covers are assumed to be overlain by a layer of rocks.

Unit man-days and machine-days are listed in Tables C-18 and C-19 on a per disposal unit basis. These tables also list the maximum quantity of waste that can be disposed per disposal unit as a function of emplacement technique.

Table C-17. Unit Man- and Machine-Day Requirements (per Disposal Cell) for Dynamic Compaction, Not Including Initial Backfilling

Basis	Ref Trench, Humid Trench	Small Trench, Humid Trench	Large Trench, Arid Trench	Small Trench, Arid Trench	Trench Ext., Arid Trench	Auger, Humid 40 grp	Slit Trench, Humid Trench	Slit Trench, Arid Trench
Max. Waste Vol. (m^3)	16,490/24,730	1090/1640	30,100/45,140	1480/2210	215	997	739	752
Emplacement	R/S	R/S	R/S	R/S	S	S	S	S
Personnel:								
· H.E. Operator	48.4	5.4	48.4	5.4	1.1	2.5	2.2	2.2
· S. Laborer	23.5	2.6	23.5	2.6	0.5	1.2	1.1	1.1
· Laborer	98.2	11.1	98.2	11.1	2.3	5.2	4.3	4.3
· Rad. Safety Tech.	33.2	3.7	33.2	3.7	0.7	1.7	1.5	1.5
· QA Technician	1.7	0.2	1.7	0.2	0.04	0.09	0.08	0.08
Equipment:								
· Bulldozer	11.8	1.3	11.8	1.3	0.3	0.6	0.5	0.5
· Loader	5.9	0.7	5.9	0.7	0.1	0.3	0.3	0.3
· Dump Truck	23.5	2.6	23.5	2.6	0.5	1.2	1.1	1.1
· Compactor	14.1	1.6	14.1	1.6	0.3	0.7	0.6	0.6
· 100-ton Crane	16.6	1.9	16.6	1.9	0.4	0.9	0.7	0.7

Table C-18. Unit (per Disposal Cell) Man- and Machine-days as Function of Disposal Technology, Reference Cover

Disposal Technology	Basis	Max Waste Vol. (m³)	Personnel				Equipment				
			H.E. Operator	S. Laborer	Laborer	Q.A. Tech.	Bulldozer	Loader	Truck	Blade	Tanker
Ref. Trench, H	trench	16,490/24,730	34.26	28.58	28.58	0.9	13.17	6.58	26.53	4.51	2.25
Small Trench, H	trench	1090/1640	3.29	3.88	3.88	0.08	1.79	0.89	3.57	0.61	0.31
Large Trench, A	trench	30,100/45,140	31.79	30.34	30.34	0.9	15.18	7.58	30.34	9.03	0
Small Trench, A	trench	1480/2210	4.31	4.11	4.11	0.1	2.06	1.03	4.11	1.22	0
Trench Extension, A	trench	215	0	0	0	0	0	0	0	0	0
Auger, H	40 grp	997	14.46	17.04	17.04	0.5	7.85	3.92	15.7	2.69	1.34
Auger, A	40 grp	6,150	19.95	18.09	18.09	0.6	9.05	4.52	18.09	5.38	0
Slit Trench, H	trench	739	1.77	2.08	2.08	0.06	0.96	0.48	1.92	0.33	0.16
Slit Trench, A	trench	752	2.32	2.21	2.21	0.07	1.11	0.55	2.21	0.66	0
Con. Slit, H	6 grp	937	8.87	11.03	11.03	0.3	5.27	2.64	10.55	0.96	0.48
Con. Slit, A	6 grp	937	10.47	11.41	11.41	0.3	5.70	2.85	11.41	1.92	0
Con. Trench, H	trench	4900	12.69	15.7	15.7	0.4	7.48	3.74	14.97	1.47	0.73
Con. Trench, A	trench	4900	15.14	16.27	16.27	0.5	8.13	4.07	16.27	2.94	0
Caissons, H	40 grp	51.2	3.30	4.11	4.11	0.1	1.97	0.99	3.94	0.34	0.17
Caissons, A	40 grp	51.2	3.87	4.24	4.24	0.1	2.12	1.07	4.24	0.68	0
Repack, H	trench	12,500	59.32	75.16	75.16	2.1	36.41	18.2	72.81	4.71	2.35
Repack, A	trench	12,500	33.16	31.66	31.66	1.0	15.82	7.92	31.66	9.42	0

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Table C-19. Unit (per Disposal Cell) Man- and Machine Days as Function of Disposal Technology, Improved Cover

Disposal Technology	Basis	Max Waste Vol. (m^3)	Personnel				Equipment					
			H.E. Operator	S. Laborer	Laborer	Q.A. Tech.	Bulldozer	Loader	Truck	Blade	Compactor	Tanker
Ref. Trench, H	trench	16,490/24,730	47.5	28.6	28.6	1.0	13.2	6.6	26.3	13.5	14.2	2.3
Small Trench, H	trench	1090/1640	6.4	3.9	3.9	0.1	1.8	0.9	3.6	1.8	1.9	0.3
Large Trench, A	trench	30,100/45,100	50.6	34.4	34.4	1.2	17.2	8.6	34.4	9.0	15.8	0
Small Trench, A	trench	1480/2210	6.9	4.7	4.7	0.2	2.3	1.2	4.7	1.2	2.1	0
Trench Extension, A	trench	215	0	0	0	0	0	0	0	0	0	0
Auger, H	40 grp	997	28.3	17.0	17.0	0.6	7.8	3.9	15.7	8.1	8.5	1.3
Auger, A	40 grp	6,150	30.2	20.5	20.5	0.7	10.2	5.1	20.5	5.4	9.4	0
Slit Trench, H	trench	739	3.2	1.9	1.9	0.07	0.9	0.4	1.7	1.0	0.9	0.2
Slit Trench, A	trench	752	3.7	2.5	2.5	0.09	1.3	0.6	2.5	0.7	1.2	0
Con. Slit, H	6 grp	937	22.2	13.6	13.6	0.5	6.6	3.3	13.2	4.8	7.6	0.5
Con. Slit, A	6 grp	937	20.4	16.2	16.2	0.5	8.1	4.1	16.2	1.9	6.3	0
Con. Trench, H	trench	4900	32.5	19.8	19.8	0.7	9.5	4.8	19.0	7.4	10.9	0.7
Con. Trench, A	trench	4900	25.1	17.6	17.6	0.6	8.8	4.4	17.6	2.9	9.0	0
Caissons, H	40 grp	51.2	8.0	5.0	5.0	0.2	2.4	1.2	4.9	1.7	2.8	0.2
Caissons, A	40 grp	51.2	6.5	4.6	4.6	0.2	2.3	1.1	4.6	0.7	2.4	0
Repack, H	trench	12,500	120.2	75.2	75.2	2.7	36.4	18.2	72.8	23.6	42.0	2.4
Repack, A	trench	12,500	52.8	35.8	35.8	1.2	17.9	9.0	35.8	9.4	16.5	0

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These estimates are based on use of heavy equipment to transport, dump and smooth cover materials. Unit equipment machine-days were based on information obtained from references 16 and 17:

Machine	Units
Bulldozer	600 yd^3/day
Loader	1200 yd^3/day
Dump Truck	300 yd^3/day
Motor Grader (Blade)	1600 yd^2/day
Compactor	500 yd^3/day
Tank Truck	3200 yd^2/day

Unit man-day requirements were estimated by assuming that heavy equipment operators were required to operate bulldozers, loaders, blades, and compactors; but that dump trucks and tank trucks may be operated by skilled laborers. General laborers are also needed and their man-day requirements are assumed to be the same as skilled laborers.

Facility and Grounds Maintenance and Radiation Survey. This factor considers a number of related activities. One activity consists of inspection, maintenance, and repair when needed of facility roads, fences, buildings, and other structures. Another activity would consist of inspection and maintenance of completed and covered disposal cells. This would include, for example, inspection for anomalies such as disposal cap subsidence holes or gopher holes, repair of same, and mowing the grass. Site surveys include inspection of buildings, facilities, structures, and grounds for fixed and removable radioactive contamination. These activities are related to the environmental monitoring program but serve a different function. They are typically performed by taking smear samples and by using hand-held instruments such as survey meters.

The above activities are projected to be a function of three factors: (1) the size of disposal facility operations, (2) the complexity of disposal facility operations, and (3) the structural stability of the disposal cells. These factors are intuitively clear. The larger the disposal facility -- i.e., the larger the annual volume of waste received -- the more extensive the need for fences, buildings, and structures. This in turn implies a need for more time spent in repair and maintenance activities. Similarly, a more highly complex operation, such as the waste repack option, implies the need for additional maintenance and repair. Finally, additional repair and maintenance would be associated with waste disposal cells filled with compressible waste such as trash, as compared with waste disposal cells containing stabilized waste such as solidified liquids.

Site radiation surveys are difficult to project, since the same types of activities would be carried out at any size facility. Company policy probably has as much influence on the extent of the survey program as external considerations such as waste volume. In any case, the extent of the survey program is assumed to be a mild function of waste volume, with the exception of the waste repack option. Due to the waste processing and additional waste handling activities performed for this option, additional surveys are projected to be performed. This is principally due to worker safety considerations.

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The assumed personnel man-day requirements are listed below, where the unit (man-day/m^3) requirements are assumed to be distributed among the listed crews: Unit requirements relate to the total cumulative volume (in m^3) of waste delivered to the site to that operational year.

Activity	Crew	Man-Day/(1000 m^3)
Maintenance of structures and grounds	1 QA technician 1 laborer 2 skilled laborers	0.2 (a)
Maintenance of disposal cells	1 QA technician 2 laborers 1 skilled laborer	unstable: 0.2 (b)(c) stable: 0.005 (b)(c)
Radiation surveys	1 QA technican 3 Radiation safety technicians	0.1 (a)

(a) For the repack disposal option, these values are assumed to be doubled.
(b) The first value is for waste disposal in a structurally unstable manner (either unstable or disposed with other unstable waste).
(c) These values are assumed to be halved for an arid disposal site.

Maintenance of disposal cells also involves use of various heavy equipment, including a loader to transport fill and cobbles and a farm tractor for grass mowing and other activities. Machine-day requirements for this equipment are listed below, where the volume unit (cubic meters) refers to the total amount of waste that had been accumulated to that year of operation. Other pieces of light equipment such as pickup trucks are assumed to be proportional to certain categories of site personnel.

Equipment	Machine-Days/(1000 m^3)	
Loader	humid unstable:	0.1
	humid stable:	0.025
	arid unstable:	0.025
	arid stable:	0
Farm tractor	humid unstable:	0.1
	humid stable:	0.1
	arid unstable:	0
	arid stable:	0

Vehicle and Equipment Maintenance. This covers personnel costs for maintaining and servicing heavy construction equipment as well as light vehicles such as pickup trucks. Costs for such activities are considered elsewhere as part of equipment operating costs. These operating costs consider costs that would be charged to a mechanic for vehicle and equipment maintenance.

Environmental Monitoring. This personnel factor covers personnel man-hours attributable to collection of environmental samples, preparation of samples for shipment to an off-site laboratory for analysis, and interpretation of results. Personnel man-hours are assumed to be a function of the number of samples analyzed per year as discussed in the environmental monitoring cost component. Personnel contributing to this factor include radiation safety technicians and quality assurance technicians as follows:

Personnel	Man-day/Monitoring Cost ($)
Radiation safety technician	0.0075
Quality assurance technician	0.0025

Quality Assurance. Personnel requirements for quality assurance technicians have been included in the personnel cost subcomponents discussed previously.

Administration

This cost component covers cost of an offsite administrative nature spent during the operational period. It consists of two subcomponents, company administration and state or compact administration, and does not include regulatory costs, which are considered elsewhere.

Company Administration. During the preoperational period, most of the engineering, design, and other work were performed at the local office which was assumed to be located in the capital city of the state in which the facility is sited. A relatively small amount of support was provided by the main company officers. During operations the local office is retained at a reduced level, while the design and other technical work is shared by the main corporate office and by the site. Costs associated with the local office are estimated at about $75,000 per year. Corporate support costs, like the number of site administration personnel, are assumed to vary depending upon waste volume. These variable costs are estimated as follows:

Annual Volume (m^3/yr)	2,000	5,000	20,000	35,000	50,000
Costs ($ x 1000)	40	100	200	350	500

State or Compact Administration. These costs are somewhat arbitrarily estimated as $100,000 per year.

Regulatory Costs

This cost component is difficult to estimate, since there may be a wide range of costs depending upon a number of factors, including whether or not the disposal site is located in an Agreement State, whether the State or the

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Federal government acts as the site landlord, or the extent to which the State becomes involved in the regulatory process. This latter factor could be based on a spectrum of considerations ranging from purely political to purely technical.

In this report, three basic subcomponents of regulatory costs are assumed. The first subcomponent covers costs associated with license renewals, the final site closure plan, inspection fees, and amendments. These costs are derived from those presented in the draft Part 61 EIS (Ref. 1), and generally occur at intervals throughout the operating life of the site. License renewal costs are assumed to occur at 5-year intervals during the operating life of the site. The first such license renewal occurs 5 years after the first year of operation. These costs include $100,000 per renewal for an environmental assessment report plus $100,000 in regulatory renewal review fees. Final site closure plan costs include an assumed $100,000 to develop the final plan plus $100,000 in regulatory review fees. These costs are assumed to be expended during the final year of site operation.

Inspection fees are assessed equally during the operational period. At an assumed two inspections per year, this totals about $40,000 per year in fees. License amendment fees are difficult to project in both costs and timing. For this report, an average cost of $15,000 is assumed to be annually budgeted for license amendment fees. This figure is somewhat arbitrary, but is based on an assumed 1 major amendment plus 10 minor amendments over a 20-year operating period.

The second subcomponent covers costs associated with the presence of an onsite State inspector. This is believed to be a likely cost component whether or not the site is located in an Agreement State. Based on the salary for a senior engineer, or equivalent, these costs are estimated as follows:

Base cost	$35,000
20% benefit	7,000
50% overhead	21,000
	$63,000

Finally, expenses resulting from confirmatory monitoring and analyses are expected. These would probably be carried out by a State government regardless of the locus of principal licensing authority, and regardless of site ownership. An annual sum of $50,000 is assumed.

Total regulatory costs therefore come to a base annual cost of $170,000 per year, which is assumed to be a constant for all waste volumes and disposal facility design mixes, plus additional costs which occur at intervals throughout the life of the site. These total $200,000 at 5-year intervals plus $200,000 on the final year of site operations.

Consulting and Studies

This cost component covers expenses associated with special hydrogeological, structural, environmental, or other studies carried out during site operations. The need for such studies is rather unpredictable on a generic basis, and so

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a constant $100,000 is assumed each year for all waste volumes and disposal technologies.

Legal Fees

Legal expenses are assumed to be constant for all waste volumes and disposal technologies. As in the final Part 61 EIS (Ref. 2), an annual sum of $150,000 is budgeted for this purpose.

Public Outreach

As in preoperational costs, an annual sum of $100,000 is assumed to be budgeted throughout the operational period.

Personnel Training

This cost component covers expenses associated with an employee training program carried out continuously during the operational period. This training program addresses areas such as the hazards and controls of radioactive materials, industrial safety, and emergency situations. The training program for a particular individual is commensurate with that individual's duties and responsibilities, and so the program is much more detailed for personnel routinely involved in waste disposal activities (laborers, equipment operators, etc.) than certain administrative personnel (clerical, shipment schedulers, etc.). The level of required training, therefore, varies considerably from one occupation to another.

In this report, a simplifying assumption is made that there are basically two levels of training. A high level of training is assumed for individuals that routinely work in the restricted area. These individuals are assumed to correspond to those discussed below ("Personnel Monitoring" cost component) who routinely participate in the personnel monitoring program. A low level of training is assumed for personnel of an administrative nature who would rarely need to enter the restricted area. Much of the training program is carried out onsite. For workers (laborers, equipment operators) and technicians, a portion of the yearly activities is automatically allotted to training and drills. This cost component therefore considers offsite training activities such as special classes.

Total annual personnel training cost is calculated here as the sum of the products of the number of appropriate individuals with the average annual training costs per individual. Thus,

$$\text{Annual Costs} = (N_b C_b) + (N_o C_o) \qquad \text{(C-13)}$$

where:

N_b = number of badged employees;
N_o = number of other employees;
C_b = annual average training costs for badged employees; and
C_o = annual average training costs for other employees.

C_b and C_o are assumed to be $1500/yr and $500/yr, respectively. For badged employees, this roughly corresponds to an average of one out-of-state training class per person. This assumes \$500 (round trip) for travel, a \$500 charge for a 5-day training course, and \$100/day per diem.

N_o is calculated as discussed above (or below) based on waste volume, while N_b is calculated based on waste volume and disposal design. Base personnel numbers as a function of waste volume are listed below.

Personnel	2,000	5,000	20,000	35,000	50,000
N_o	3	4	8	13	18
N_b	8	9	12	16	18

For N_b, additional personnel are assumed based on calculations performed for the above personnel cost component. These personnel include heavy equipment operators, laborers, skilled laborers, surveyors, QA technicians, and radiation safety technicians.

Disposal Cell Materials

This cost component considers costs for specific disposal cell materials which are assumed to be purchased and obtained from off the disposal site. It does not include personnel costs for disposal cell construction, heavy equipment costs, nor costs for heavy equipment lubrication, fueling, and maintenance. It also does not include costs for quality assurance testing and inspection, nor costs for soil or topsoil obtained from an onsite borrow. Such material can be obtained onsite without cost.

To develop these costs, unit quantities of specific materials are developed for each disposal technology and operational variation. Given the scope of this report, only the principal material types are identified. Minor components such as form work tie bolts and expansion joints for concrete disposal methods are not considered in detail since they are not believed to significantly contribute to material costs. Following this, unit material costs are presented. This is finally followed by material costs tabulated on a per disposal unit basis.

Unit material quantities are listed in Table C-20. Each disposal technology is listed along with design (arid or humid) and operational (cover, emplacement, backfill) variations. In the table, standpipes are given in units of length, which is the product of the number of standpipes and the assumed length per standpipe. Other unit quantities are given in units of area (formwork, asphalt, and sealant for concrete disposal technologies). Other cost assumptions such as average thicknesses are assumed below. Rebar requirements are expressed in terms of length and are based on general assumption of a double layer of cross-hatched rebar in concrete walls, lids, and floors. One layer of rebar is assumed to have 6-in. (0.15 m) spacing while the second layer is assumed to have 12-in. (0.3 m) spacing.

Table C-20. Unit Material Quantities for Disposal Technology and Operational Variations

Disposal Technology: · Emplacement · Basis · Waste Volume (m³)	Reference Trench, Humid Ran/Stk(a) Per trench 16,490/24,730	Small Trench, Humid Ran/Stk(a) Per trench 1090/1640	Large Trench, Arid Ran/Stk(a) Per trench 30,100/45,140	Small Trench, Arid Ran/Stk(a) Per trench 1480/2210	Trench Extension, Arid Stk(b) Per extension 215	Shallow Auger, Humid Stk Per group (c) 997	Deep Auger, Arid Stk Per group (c) 6,150	Slit Trench, Humid Stk Per trench 739	Slit Trench, Arid Stk Per trench 752
Disposal Cell:									
· Sand (m^3)	1367	112	0	0	0	0	0	0	0
· Gravel (m^3)	67	24	0	0	0	111	0	18	0
· Crushed stone (m^3)	42	17	42	17	0	114	114	10	10
· Standpipes (m)	27	14	0	0	0	280	0	14	0
· Access Boxes (m)	3	2	0	0	0	40	0	2	0
· Markers (#)	4	4	4	4	0	40	40	1	1
· Monuments (#)	1	1	1	1	0	1	1	1	1
· Concrete (m^3)	0	0	0	0	0	0	0	0	0
· Rebar (m)	0	0	0	0	0	0	0	0	0
· Formwork (m^2)	0	0	0	0	0	0	0	0	0
· Asphalt (m^2)	0	0	0	0	0	0	0	0	0
· Sealant (m^2)	0	0	0	0	0	0	0	0	0
Reference Cover:									
· Seed (m^2)	6040	819	0	0	0	3600	0	441	0
· Cobbles (m^3)	0	0	920	125	0	0	549	0	67
Improved Cover:									
· Seed (m^2)	6040	819	0	0	0	3600	0	441	0
· Cobbles (m^3)	0	0	1840	250	0	0	1100	0	0
· Clay (m^3)	3020	410	0	0	0	1800	0	221	0
· Sand (m^3)	0	0	0	0	0	0	0	0	0
· Gravel (m^3)	0	0	0	0	0	0	0	0	0
Backfill: (g)									
· Sand (m^3)	21,840/13,590	1670/1130	35,440/20,400	2060/1320	181(d)	614	2330	471	475
· Grout (m^3)	16,490/8,240	1090/545	30,100/15,050	1480/738	71.6(d)	331	2050	246	251

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Table C-20. Unit Material Quantities for Disposal Technology and Operational Variations (Continued)

Disposal Technology: · Emplacement · Basis · Waste Volume (m³)	Concrete Slit Trench, Humid Stk Per group(e) 937	Concrete Slit Trench, Arid Stk Per group(e) 937	Concrete Trench, Humid Stk Per trench 4900	Concrete Trench, Arid Stk Per trench 4900	Caissons, Humid Stk Per group (f) 51.2	Caissons, Arid Stk Per group (f) 51.2	Repack, Humid Stk Per trench 12,500	Repack, Arid Stk Per trench 12,500
Disposal Cell:								
· Sand (m³)	90	90	311	311	160	165	794	749
· Gravel (m³)	213	213	752	752	0	0	1499	1499
· Crushed Stone (m³)	15	15	15	15	11	11	33	34
· Standpipes (m)	84	42	56	42	14	14	7	9
· Access Boxes (#)	12	6	8	6	2	2	1	1
· Markers (#)	12	12	4	4	40	40	4	4
· Monuments (#)	1	1	1	1	1	1	1	1
· Concrete (m³)	812	812	1835	1835	102	102	1298	1298
· Rebar (m)	54,640	54,640	101,500	101,500	14,420	14,420	86,510	86,510
· Formwork (m²)	4123	4123	4885	4885	1194	1194	190	190
· Asphalt (m²)	969	0	985	0	433	0	0	0
· Sealant (m²)	2172	2170	2179	2179	738	738	0	0
Reference Cover:								
· Seed (m²)	1290	0	1960	0	459	0	6300	0
· Cobbles (m³)	0	197	0	299	0	70	0	960
Improved Cover:								
· Seed (m²)	1290	0	1960	0	459	0	6300	0
· Cobbles (m³)	0	393	0	597	0	140	0	1920
· Clay (m³)	645	0	980	0	229	0	3150	0
· Sand (m³)	339	0	442	0	134	0	1,890	0
· Gravel (m³)	291	0	296	0	130	0	1,890	0
Backfill: (g)								
· Sand (m³)	312	312	1355	1355	7.61	7.61	6020	6020
· Grout (m³)	312	312	1355	1355	7.61	7.61	6020	6020

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Table C-20. (continued)

(a) Waste may, at the option of the code user, be disposed in either a random or stacked manner. Depending upon the option chosen, some material quantities vary on a trench basis as indicated.

(b) Waste is always assumed to be stacked in the slit trench extension. Waste placed within the larger trench above the slit trench may be disposed in either a stacked or random manner.

(c) Per group of 40 3-m diameter auger holes.

(d) This volume refers only to the slit trench extension at the bottom of the larger trench.

(e) Per group of 6 concrete slit trenches.

(f) Per group of 40 caissons.

(g) Volumes assuming maximum waste volumes per disposal cell as listed above.

Assumed unit material costs are listed below:

Material	Cost ($)	Unit
Sand	4	m^3
Gravel	4	m^3
Crushed Stone	4	m^3
Standpipes	7	m
Access Boxes	140	each
Markers	20	each
Monuments	200	each
Concrete	45	m^3
Rebar	1.3	m
Formwork	10	m^2
Asphalt	6.5	m^2
Sealant	1.3	m^2
Seed	0.18	m^2
Cobbles	5	m^3
Clay	5	m^3
Grout	45	m^3

These unit costs are estimated generally based upon information obtained from references 16-18. Sand and gravel costs are based on an assumed use of bank sand and bank run gravel. Standpipe costs are based on an assumed 4-in diameter PVC piping. Access box costs are based on an assumed use of 24-in square concrete frames with covers (Ref. 17). Costs are approximated based on use of a 6 ft^2 slab of 4-in thick gray domestic granite (Ref. 17) onto which a metal plate is affixed which describes the disposal cell contents.

Concrete costs are given for materials only, since concrete is assumed to be made onsite rather than offsite. This is due to the probable stringent quality assurance and testing criteria on the concrete (and materials) chemical and physical composition. Ready mix concrete costs about $45/CY for 3000 psi concrete, while costs for materials (cement, sand, aggregate) run at about $30 per CY of mixed 3000 psi concrete (Ref. 17). This price is inflated slightly (about 15%) to account for additional chemical agents such as air entraining or bonding agents, and also to account for additional costs (including paperwork costs) required to test and track the quality and composition of the materials. An equivalent materials cost of $45 per CY of mixed concrete is assumed.

Rebar costs are based on an assumed 3/4-in diameter bar and a price of about $550/ton. Formwork costs are an estimate from a number of sources assuming use of plywood. Asphalt costs are based on approximate material costs per square yard for installing a 3-in parking area (Ref. 17). Sealant costs are based on two sprayed-on coats of asphalt coating (Ref. 17).

Seed costs are based on use of a hydraulically applied combination of seed, fertilizer, and wood fiber mulch at a material cost of about $0.15/SY. Grout costs are assumed to be the same as concrete costs.

Finally, minimum unit costs are listed in Table C-21 for each disposal technology variation and operational variation that results in acquisition of offsite materials. These unit costs were calculated by multiplying unit material costs by the information in Table C-18, and do not include backfill costs (grout or sand/gravel). Costs for backfill materials are calculated separately, since possible regulatory restrictions on disposal of certain waste classes could result in less efficient use of some disposal cells. This problem was noted above for the Backfilling subcomponent of the Salaries cost component. As before, it is useful to separate the backfill emplaced in the interstitial spaces between waste packages ("waste backfill") from the backfill emplaced between the disposed waste and the top of the disposal cell ("plug backfill"). (See Figure C.15.) The waste backfill (WB) within a disposal cell is given by:

$$WB = \frac{(1- EMP) * WD}{EMP} \qquad \text{(C-12)}$$

where

WB = waste backfill volume (m^3);
EMP = emplacement efficiency; and
WD = waste volume emplaced within a disposal cell. (WD may be the sum of waste volumes contributed by a number of waste classes.)

The plug backfill volume (PB) is given as the empty volume of the disposal cell (that is the volume of the completed disposal cell prior to waste emplacement) minus the quality WD/EMP.

Somewhat different cost assumptions are made depending upon whether engineered or nonengineered disposal methods are used. For concrete slit trenches, concrete trenches, and caissons, backfill material costs are determined by multiplying unit material costs (\$45/m^3 for grout and \$4/m^3 for sand/gravel) by the quantity (WB + PB). For all other disposal technologies, backfill material costs are determined by multiplying unit costs by the quantity WB. This is done to minimize expenses, particularly in the case of grout.

Other Materials

This includes materials (other than utilities, fuel oil, equipment replacement, etc.) used in the repack disposal option. Specifically, this cost component includes individual material costs for the disposal blocs plus additional costs for the grout used to fill the voids within the blocs after filling the blocs with waste. Repack material costs associated with utilities, fuel, equipment maintenance, personnel requirements, and disposal cell materials are considered elsewhere.

Bloc costs are assumed to be approximately \$900 per empty bloc. Based on the assumptions presented in Section C.3.1.12, an average bloc is assumed to contain approximately 1.3 m^3 of grout. At an average material cost for grout of about \$45/m^3, this results in a grout cost per bloc of about \$60. The total cost with grout fill is thus about \$960 per bloc. Each block can contain an average of about 4.1 m^3 of waste.

Environmental Monitoring

This cost component covers sample analysis costs associated with isotope-specific radiation surveys and environmental monitoring. It does not include personnel

Table C-21 . Unit (per Disposal Cell) Materials Costs for Disposal Technologies and Operational Variations

Disposal Techology	Ref. Trench, Humid	Small Trench, Humid	Large Trench, Arid	Small Trench, Arid	Trench Extension, Arid	Auger, Humid	Auger, Arid	Slit Trench, Humid	Slit Trench, Arid
Basis	trench	trench	trench	trench	lrg. trench	40 grp	40 grp	trench	trench
Vol.(m^3)	16,490/24,370	1090/1640	30,100/45,140	1480/2210	215	997	6,150	739	752
Emp.	R/S	R/S	R/S	R/S	S	S	S	S	S
Disposal Cell	6792	1272	447	349	0	9,459	1455	712	260
Ref. Cover	1087	147	4602	624	0	648	2743	79	336
Imp. Cover	16,185	2195	9203	1248	0	9648	5486	1182	672

Disposal Technology	Con. Slit, Humid	Con. Slit, Arid	Con. Trench, Humid	Con. Trench, Arid	Caissons, Humid	Caissons, Arid	Repack, Humid	Repack, Arid
Basis	6 grp	6 grp	trench	trench	40 grp	40 grp	trench	trench
Vol.(m^3)	937	937	4900	4900	51.2	51.2	12,500	12,500
Emp.	S	S	S	S	S	S	S	S
Disposal Cell	161,899	154,469	278,755	271,974	41,113	38,300	182,341	182,343
Ref. Cover	232	983	353	1494	83	349	1134	349
Imp. Cover	5975	1666	8203	2987	2203	699	32,004	9601

costs nor surveys performed with ordinary survey instruments. Nor does it include costs associated with beta-gamma surveys of removable contamination (e.g., swipes of incoming and exiting delivery vehicles). Rather, it consists of costs associated with sample analyses which require more complicated procedures such as radiochemical separation techniques or equipment such as GeLi scintillation detectors, multichannel analyzers, and data processors. Such sample analyses are assumed to be performed offsite at a commercial laboratory.

It is true that a very large site may have a complete laboratory which can perform such analyses onsite; in fact, one such laboratory is functioning at an existing operating disposal site. However, this site receives approximately half of the civilian waste annually generated in the United States, and the operators of the site also perform a number of business activities (transportation, waste treatment, solidification, and packaging services, decontamination services, etc.) which are not directly related to waste disposal operations. This onsite environmental laboratory handles considerably more business than would be the case if it performed sample analyses connected only with the disposal site itself.

The various types of environmental monitoring activities which make up this cost component are listed in Table C-22. In general, such costs from a strictly technical viewpoint would be expected to be a weak function of site capacity (waste volume received), and also a function of disposal technology, depending upon the type of sample collected and analyzed. However, such environmental monitoring activities are also carried out from a public acceptance viewpoint -- to demonstrate to the surrounding population that the disposal site does not represent a hazard -- and it is difficult to factor this consideration into the analysis. Such considerations may be totally site-specific and are therefore difficult to account for in a generic analysis. Another consideration is that more monitoring activities would be expected at a humid site than at an arid site.

TLD badges are used to help assess the potential offsite dose due to direct radiation and also the radiation environment in a particular operational area. A given number (say 30-50) would be assumed for any site size or design, and since the analysis costs are relatively inexpensive, any variation due to site size or design would be insignificant in terms of costs. Offsite sample analysis of wells, etc., would also be performed independent of the site size or design. The total number of boundary wells would be expected to vary due to site size -- i.e., a larger site implies more wells -- but this variation is not believed to be strong.

Sampling costs which are believed to be the strongest function of disposal technology and waste volume include those associated with: (1) particulate air filters, and (2) onsite monitoring wells and disposal cell sumps. Additional air filter sample analyses would be expected for disposal technologies for which there is a potential for airborne dispersion (e.g., extreme compaction alternative), and for disposal alternatives in which more than one disposal cell would be in operation at any given time (e.g., segregated waste disposal). Analysis of onsite monitoring wells would be performed periodically while analysis of disposal cell sumps would take place only if water were determined to be present in a sump. Thus, such costs are a function of the total number of wells and sumps surveyed (a function of waste volume) as well as the disposal design and the site environment. An improved disposal design allowing reduced amounts of

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Table C-22. Types of Environmental Monitoring Activities

Sample Description	Type	Media	Typical Frequency of Analysis	Typical Type of Analysis	Assumed Variation
External Gamma	Continuous	TLD	Quarterly	Exposure	Mostly Constant
Atmosphere	Continuous	Particulate Filter Particulate Filter Particulate Filter	Daily Weekly Weekly	Gross Beta-Gamma Gamma Isotopic I-131	Disposal Design
Soil and Vegetation	Grab	Soil and Vegetation	Quarterly	Gross Beta-Gamma Gamma Isotopic Gross Alpha Tritium	Waste Volume, Weakly
Offsite Wells	Grab	H_2O	Semiannual	Gamma Isotopic Gross Alpha Tritium	Mostly Constant
Site Boundary Wells	Grab	H_2O	Semiannual	Gamma Isotopic Gross Alpha Tritium	Waste Volume, Weakly
Disposal Area Wells	Grab	H_2O	Quarterly	Gamma Isotopic Gross Alpha Tritium	Waste Volume, Weakly
Disposal Cell Sumps	Grab	H_2O as Observed	Sample Monthly, Analyze as Needed	Gamma Isotopic Gross Alpha Tritium	Disposal Design and Waste Volume

percolation would imply fewer samples, as would a site location in an arid environment. Conversely, a humid site environment implies additional samples.

Another consideration is that costs for analysis of many types of samples are a function of waste inventory. For example, there will be few wells or sumps to survey at the beginning of disposal site life and possibly several hundred at the end of disposal site life. This growth in monitoring activities as a function of time would also be the case to a lesser extent with onsite and boundary well samples and soil and vegetation samples.

Considering all the above, environmental monitoring costs are assumed to vary as follows. A base annual cost is first assumed which varies somewhat depending on waste volume and site environmental conditions. This variation is given below.

Base Environmental Monitoring Costs ($ x 1000)

Site Environment	Annual Waste Volume (m^3/yr) 2,000	25,000	50,000+
arid	40	60	80
humid	60	80	100

Additional sample costs are assumed for particulate air filter sample analysis and for disposal cell sump analysis. For air monitors, an annual sample cost of $14k is assumed per air monitor. A minimum of three air monitors is assumed, and depending upon the disposal technology option considered, additional air monitors may be used as listed below:

($ x 1000)

Base Costs: 42
Chemical waste segregation: +14
Unstable waste segregation: +14
Disposal of Class D waste: +14
Extreme stabilization: +70

For onsite wells and sumps, annual costs are estimated as follows:

$$\text{Annual costs} = V_c \times C_u \qquad \text{(C-14)}$$

where

V_c = total waste volume disposed up to the operational year considered; and
C_u = unit analysis cost per waste volume disposed.

In this report, C_u varies depending on the disposal technology (stable or unstable) and the site environment. The following relationship is assumed:

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($\$/m^3$)

Site Environment	Stable	Unstable
arid	.003	.03
humid	.03	.3

Personnel Monitoring

Personnel monitoring costs are determined in a similar manner as those in the final Part 61 EIS (Ref. 2). First, personnel are identified which are assumed to require reasonably routine access to the disposal facility restricted area. These personnel are assumed to correspond to those discussed above under the Personnel Training cost component as receiving a high level of radiation and industrial safety training. Second, a yearly personnel monitoring cost is determined per individual based on annual costs for 12 TLD analyses, 1 whole body scan, and 2 bioassays. This is assumed to be about $600/yr. Last, the individual costs are multiplied by the total number of badged personnel. These personnel (N_b) are calculated as discussed above for the training cost component.

Equipment Replacement

This cost component can be divided into two general groups of costs. The first involves costs associated with replacement of heavy equipment and site vehicles (4-wheel drive, pickup trucks, corporate limos). The second involves costs associated with replacement of health physics and other light equipment such as office equipment and furniture.

The assumed replacement schedule for heavy equipment and other site vehicles has been provided in Tables C-5 and C-6, along with the assumed replacement cost per unit. The actual replacement costs for a specific site are a function of the original type and number of units purchased. For example, assuming that the original preoperational cost component considered the purchase of 5 pickup trucks, then the replacement requirements would be for 5 pickup trucks every 5 years, at a cost of 5 x $10,000 = $50,000 in 1984 dollars. For tax purposes, depreciation on this equipment group is assumed for each unit over the replacement interval. No salvage value at the end of the replacement period is assumed. If the remaining operating life of the disposal facility exceeds the replacement period for a given piece of equipment, then the unit is depreciated over the remaining operating life rather than the replacement period.

Replacement of health physics, office, and other light equipment is more difficult to estimate. In this report, a 10-year equipment replacement schedule is assumed.

Miscellaneous Expenses

This cost component covers a variety of small miscellaneous costs associated with operation of a disposal facility. Items which are included under this cost component include:

- communications;
- travel (e.g., to technical seminars);
- temporary equipment rental;

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- miscellaneous supplies and expenses (e.g., word processing rental, cleaning materials, etc.); and
- personnel management costs (e.g., moving expenses).

Given the rather general nature of this cost component, it cannot be precisely estimated. Rather, a given lump sum is assumed to be budgeted each year which is assumed to be a function of waste volume delivery rate. Assumed annual costs are listed below.

Waste Volume (m^3/yr)	Miscellaneous Costs ($ x 1000)
< 10,000	150
10,000 < 20,000	200
20,000 < 30,000	250
30,000 < 40,000	300
40,000 < 50,000	350
50,000+	400

Utilities

This cost component covers use of utilities during the facility operational period. It is assumed to vary depending on the waste volume with some additional variation depending upon use of some specific disposal technologies. The principal utility cost is assumed to be that for electricity, since water is provided by an onsite well installed by the site operator. Fuel oil for equipment is considered as part of equipment operating costs.

Base costs are given below as a function of waste volume:

Waste Volume (m^3/yr)	Costs ($ x 1000)
< 10,000	30
10,000 < 20,000	40
20,000 < 30,000	50
30,000 < 40,000	60
40,000 < 50,000	70
50,000+	80

Additional costs are assessed for operation of a cement plant as well as use of the waste repack disposal option. In the former case, additional utility costs are estimated as $0.10 per cubic meter of concrete or grout produced. For the repack disposal option, additional utility costs are estimated as $5 per m^3 of waste input to the disposal facility.

Heavy Equipment Operating Costs

This cost component covers costs for operation and maintenance of site vehicles and heavy equipment. It includes fuel, oil, lubrication, normal expendables for the equipment, and a percentage of a mechanic's wages chargeable to maintenance. It is estimated as a function of vehicle and equipment type based on yearly operational hours. Unit operating costs are estimated based on reference 17, and are summarized in Table C-23 for the principal types of equipment utilized. Operating costs for the cement plant are estimated at about $1 per m^3 of cement or grout produced. For pickup trucks, 4WD vehicles, and sedans, annual operating costs are estimated as approximately $2,500 per vehicle.

Table C-23. Equipment Operating Costs ($/Machine-Day)

Equipment	$/Machine-Day	Equipment	$/Machine-Day
Bulldozer	210	Auger rig	200
Front end loader	140	Paving machine	85
Dump truck	100	Tandem roller	40
Pan scraper	280	Compactor	100
Motor grader	130	Forklift, large	60
Backhoe	45	Farm tractor	25
40-ton crane	115	Hand tamper	3
100-ton crane	250	Cement truck	25
Forklift, small	50	Mobile cem. pump with boom	220
Water truck	70		

Maintenance

This cost component accounts for replacement costs associated with routine upkeep of site grounds, buildings, and other site structures such as roads, fencing, lighting, etc. These costs are estimated to the 1% of the capital outlay for these grounds, buildings, facilities, and other structures per year. These costs are for "hardware" only. Personnel requirements for site upkeep and repair are considered elsewhere.

Insurance

A flat rate of $150,000/yr is assumed for this cost component, which covers nuclear liability insurance costs. Personnel insurance costs and benefits are included in the personnel cost overhead.

Quality Assurance and Compliance Testing

This cost component covers costs, other than personnel costs, associated with performing quality assurance and compliance testing. Personnel costs required to perform testing have been considered earlier. As for the environmental monitoring cost component, this cost component is concerned with structural sampling and testing that requires use of offsite laboratories and/or special expertise.

To date, there have been few or no regulatory requirements for structural stability testing for waste disposal cells. However, some testing would be expected for any new disposal facilities incorporating shallow land burial or some other form of nonengineered disposal method. Significantly additional test requirements would be expected for engineered methods. Guidance on the types of tests that would be needed for waste disposal still needs to be developed on a systematic basis. Pending the development of this guidance, there is little that can be done in accurately predicting costs. However, it is possible to briefly list some of the testings that could be performed, and to project some very preliminary estimates of costs.

For soils, some of the tests that could be performed are listed below along with typical (1984) costs (Ref. 17):

Test	Costs (each)
Attenberg limits	$30
Hydrometer analysis and specific gravity	50
Sieve analysis	40
Consolidation test	400
Density of undisturbed sample	22
Moisture content	7
Permeability	65
Proctor compaction	100
Triaxial shear test	150-280
Direct shear test	95-260

For concrete, some of the tests that could be performed include (Ref. 17):

Test	Cost (each)	Test	Cost (each)
Aggregates, abrasion	$60	Compressive strength cylinders (test only)	$8-17
Absorption	25		
Petrographic analysis	440	Compressive strength, cores	20
Specific gravity	25	Drying shrinkage at 28 days	240
Sieve analysis	35-65	Flexural test beams	30
Sulfate soundness	66	Trial mix batches	150
Weight per cubic foot	15	Modulus of elasticity	120
Cement, physical tests	250	Tensile test, cylinder	50
Cement, chemical tests	300	Water/cement ratio (3 batches)	450

For this report some very approximate estimates are made and presented. For nonengineered disposal methods, tests would be relatively simple and would probably be limited to chemical and structural characteristics of soil.

For engineered disposal methods, tests would be more extensive and would include physical and chemical characteristics of soil, physical and chemical characteristics of structural materials (water, cement, aggregate, rebar, etc.), and the stability of the completed structures. Assumed costs are listed in Table C-24 as a function of disposal method:

Contingency

Similarly, preoperational costs for this cost component are calculated as 20% of all of the above cost components.

Table C-24. Quality Assurance and Compliance Test Costs

Disp. Tech.	Basis	Cost($)	Disp. Tech	Basis	Cost ($)
Ref. trench, H	trench	500	Con. slit, H	6 grp	2500
Small trench, H	trench	200	Con. slit, A	6 grp	2500
Large trench, A	trench	500	Con. trench, H	trench	2500
Small trench, A	trench	200	Con. trench, A	trench	2500
Trench ext., A	trench	100	Caissons, H	40 grp	2500
Auger, H	40 grp	300	Caissons, A	40 grp	2500
Auger, A	40 grp	200	Repack, H	trench	3000*
Slit trench, H	trench	200	Repack, A	trench	3000*
Slit trench, A	trench	100			

*Includes periodic testing of blocs.

C.5 POSTOPERATIONAL COSTS

This chapter presents base postoperational costs as a function of waste volume, site environment, and disposal technology. Three types of postoperational costs are considered: closure, surveillance, and institutional control. Post operational costs are assumed to be funded by a surcharge (\$ per m^3 of waste) on waste received at the disposal facility. Money collected from this surcharge is assumed to be placed by the operator into an interest-bearing investment fund.

Costs calculated here are given in terms of constant (1984) dollars. The time value of these costs is considered in Chapter C.6.

C.5.1 Closure Costs

For this report, only one closure alternative is considered in detail: namely minimum closure activities. The disposal cells have been "closed" in progress, so that only decontamination and demolition of site structures and buildings is required. Contaminated material is disposed on site. (The code user also has the option to override any assumed costs by user-input costs.) The closure activities -- decontamination and decommissioning, backfilling and covering -- are essentially assumed to be carried out over the course of 1 year. (Additional time may be assumed by the code user.) At least 5 additional years are assumed to be needed to verify that closure has been correctly carried out, to allow newly seeded and contoured areas to become established, and so forth. The costs associated with this surveillance time period are addressed in Section C.5.2.

To determine costs, the first requirement is to determine the total amount of decommissioning material that would be generated, of which a portion could be contaminated with radioactivity and thus require onsite disposal as radioactive waste. The particular disposal technology used for disposal of the contaminated decommissioning waste is assumed to be that used for Class A (unstable) waste. This is because only small quantities of radioactive contamination are expected to be involved. Second, the individual cost components contributing to total closure costs are presented. Third, the assumed additions to operating costs due to closure surety mechanisms are addressed.

C.5.1.1 Decommissioning Waste Volumes

At least six site buildings are assumed for all disposal technologies. Three of these six buildings -- i.e., the administration building, the health physics/security building, and the site warehouse -- are located in the administrative area of the site and should be free of contamination. The administrative building and the warehouse are dismantled and sold for salvage. The health physics/security building is left standing for use by the site owner during the institutional control period. Of the remaining three site buildings, only the waste activities building is expected to have appreciable levels of contamination. This building is decontaminated to the extent practical and demolished, as is the site garage and the site shed.

Depending upon the disposal technology considered, two additional structures may be located at the disposal facility. One structure is a concrete batch plant. Although it will probably be located in the operational area, it is unlikely to be contaminated since it does not process radioactive material. This structure is removed from the site and either scrapped or sold for whatever salvage value that can be obtained. The second structure is the WPRB in the event that the repack option is assumed. Demolition of this structure is projected to generated contaminated as well as uncontaminated waste.

Costs for demolition are very difficult to generically estimate since a number of unpredictable factors are involved. These include the ease with which buildings and structures can be demolished, the quantities of both contaminated and noncontaminated material, the quantities of particular radionuclides, the waste handling requirements (e.g., contact or remote), and the salvage value of various materials. Such factors are not necessarily a function of waste volume or disposal technology. For example, the amount of radioactive contamination directly influences the ease in which a building can be demolished, the contaminated waste volume, and the possible salvage value. However, such contamination is not necessarily a function of waste throughput to the site. Unquantifiable factors such as company philosophy and operating practices, possible accidents or spills, and facility design are believed to be more important.

Therefore, contaminated waste volumes are assumed to be constant for all waste volumes received for disposal and all disposal technologies except the repack option. Based on the draft EIS (Ref. 1), contaminated decommissioning waste volumes for most applications are therefore assumed to be 1130 m^3 (40,000 ft^3). For the repack option, contaminated decommissioning waste volumes are assumed to be about three times as much, or 3400 m^3 (120,000 ft^3). This increased volume is due to the significant waste processing activities which increase the potential for building contamination. In either case, contaminated decommissioning waste is assumed to be generated in a structurally unstable form as defined by 10 CFR 61.56, is assumed to be Class A waste, and is assumed for purposes of waste handling to be "regular care" waste.

C.5.1.2 Closure Cost Components

The principal cost components contributing to site closure costs are listed below.

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- Nonradiological demolition
- Salaries
- Administration
- Regulatory costs
- Consulting and studies
- Legal fees
- Public outreach
- Disposal cell materials
- Environmental monitoring
- Personnel monitoring
- Miscellaneous expenses
- Utilities
- Equipment expenses
- Insurance
- QA and compliance testing
- Contingency

The above cost components are very similar to those for operational costs except for the additional nonradiological demolition cost component. Each cost component is very briefly discussed below.

Nonradiological Demolition

This cost component covers costs associated with dismantlement of the administration building and warehouse. Since these buildings should be free of contamination, dismantlement is assumed to be contracted to an outside firm. A constant cost of $100,000 is assumed for all waste volume and disposal technology mixes.

Salaries

Salaries include those for site administration, support, and workers. In a similar manner as for operational costs, base assumptions are made for administration and some support personnel. For other support personnel (radiation safety technicians, QA technicians, and workers) personnel requirements are calculated by identifying the principal factors contributing to personnel requirements, estimating the individual personnel man-day requirements for each factor, and summing the results. The personnel number per job category is then determined by dividing by the product of the worker efficiency factor (75%) and the number of realistic working days per year (220 days).

Base personnel costs are taken to be as follows:

Personnel	Costs ($ x 1000)
1 Site Manager	45
1 Foreman	28
1 Radiation Safety Officer	35
1 QA and Safety Supervisor	30
2 Security	36
	174
30% benefit	+ 52.2
	$226.2

For radiation safety technicians and QA technicians, the principal factors contributing to personnel estimates are taken to be as follows:

- Disposal cell construction
- Cement plant operation
- Waste processing
- Waste placement and backfilling
- Covering
- Facility and grounds maintenance and radiation surveys
- Environmental monitoring
- Quality assurance

Workers included in the calculations include the following:

Worker	Salary ($ x 1000)
Heavy equipment operator	25
Unskilled laborer	15
Skilled Laborer	25
Surveyor	60
QA technician	25
Radiation safety technician	25

Total annual salaries thus determined are then added to those for administrative and support personnel as given above. These costs are then increased assuming a 30% benefit.

Disposal cell construction. The particular disposal technology used is assumed to be the same as that for Class A (unstable) waste. A constant decommissioning waste volume is assumed -- 3400 m^3 for the repack option and 1130 m^3 for all other disposal technologies -- and construction requirements are prorated to the available disposal cell waste volume using the data in Table C-12.

Cement plant operation. Personnel requirements vary based on the disposal technology. Cement plant operation is based on: (1) concrete requirements, if any, for disposal cell construction and (2) grout backfill requirements, if the option is selected.

Waste processing. This cost factor considers personnel and machine requirements for waste packaging into blocs in the event that the repack option is chosen for Class A (unstable) waste disposal. As a very rough approximation, these requirements are calculated for 3400 m^3 of waste as if the WPRB was operating normally.

Waste emplacement and backfilling. As before, personnel and machine requirements are separated into those for waste emplacement and backfilling.

Waste emplacement requirements are estimated for all but the repack option based on an assumed use of drums for waste disposal. Assuming 1130/0.2 = 5,620 drums at a regular care level, this comes to:

	Ref. Trench, Small Humid Trench, and Small Arid Trench		All Other Disposal Methods Except Repack Option	
	Ran	Stk	Ran	Stk
Total man- and machine-minutes	16,950	67,800	22,600	73,450

The distribution among personnel and machinery is given by Table C-14.

For the repack option, 830 blocs are needed, resulting in total emplacement requirements of 41,500 man-minutes plus 41,500 machine-minutes. The distribution among personnel and machinery is given in Section C.4.3.

Backfilling requirements are dependent upon disposal technology. Personnel and machine requirements are calculated in a similar manner as that in Section C.4.3.

Covering. This factor depends upon the disposal technology and closure alternative chosen. Requirements are volume prorated to Tables C-17 through C-19.

Facility and grounds maintenance and radiation survey. These personnel requirements are assumed to be the same as those in Section C.4.3.1 under the "salaries" cost component. The waste volume used for the calculations is assumed to be the total cumulative volume received over site operation.

Environmental monitoring. These personnel requirements are assumed to be the same as those in Section C.4.3.1 under the "salaries" cost component.

Quality assurance. Personnel requirements for this cost component are included in other Personnel subcomponents as discussed above.

Administration

Corporate administration costs consist of those associated with the local office as well as company support. The local office is retained during the closure period given the importance of closure in the overall life of the site. A cost of $75,000 is therefore assumed consistent with that assumed for the operational periods. Again consistent with the approach taken for operational costs, corporate costs are assumed to vary depending upon the amount of waste disposed during closure activities. These costs are assumed to be $75,000 for disposal of decommissioning waste by the repack alternative, and $25,000 for all other alternatives.

State or compact administration costs are increased somewhat during the closure period due to the importance of closure activities. These are assumed to be $150,000.

Regulatory Costs

Based on information obtained from Section C.4.3, this cost is taken to be $135,000/yr. This includes one major inspection, two minor amendments to the site license, the annual salary for an onsite inspector, and costs for independent confirmatory monitoring and sampling.

Consulting and studies

These costs are assumed to be the same as those during site operations, or \$100,000/yr.

Legal fees

These costs are assumed to be the same as those during site operations, or \$150,000/yr.

Public outreach

These costs are assumed to be the same as for those during operational costs, or \$100,000/yr.

Disposal Cell Materials

The waste disposed is presumed to be all Class A waste, and all of a regular care handling level. Given this, there should be no or very little areal density limitation considerations. Thus, disposal cell material costs are assumed to be directly proportional to the volume of contaminated demolition waste disposed: 3400 m^3 for the repack option and 1130 m^3 for all other disposal technologies. Disposal cell material costs are obtained by multiplying the appropriate waste volume by the cost information contained in Table C-21. The particular disposal technology and set of operational practices utilized correspond to those selected by the user for Class A (unstable) waste disposal.

Environmental Monitoring

Environmental monitoring costs are assumed to be the same as those for the final year of site operations.

Personnel Monitoring

As for operational costs, these costs are taken to be \$600/yr times the total number of closure personnel. These are calculated using the procedures outlined above for the salaries cost component.

Miscellaneous Expenses

These costs include a variety of small miscellaneous costs associated with facility closure. These include such items as communications, travel, temporary equipment rental, miscellaneous supplies and expenses, and personnel management costs. Based on assumptions for operating costs, this cost component is assumed to total \$150,000.

Utilities

This cost component is expected to be only slightly different depending upon the particular mix of disposal technology, environment, and waste volume. For simplicity, then, this cost component is assumed to be a constant \$30,000 in all cases.

Equipment Expenses

This cost component is calculated in a somewhat similar manner as in Section C.4.4.3. The total machine-day requirements for heavy equipment are determined in the same manner as during the operational period. These machine-day requirements are then multiplied by the machine-day costs listed in Table C-25. These machine-day costs are the sum of daily operating and leasing costs. Daily leasing costs are calculated as one-fifth of weekly costs as obtained from references 16-18. An 80% efficiency factor is also included in the calculations. Leasing costs are included since the heavy machinery used during the operational period would theoretically be depreciated. It also better reflects a more conservative situation in which someone other than the site operator performs closure.

Insurance

This cost component is taken to be the same as that for operational costs, or $150,000/yr.

QA and Compliance Testing

This cost component is determined using the information in Table C-22, and prorating unit costs given in Table C-22 to the volume of waste to be disposed as part of closure. This waste volume is assumed to be 1130 m^3 for all Class A (unstable) disposal methods except for the repack option, in which case the waste volume is 3,400 m^3.

Contingency

As in the preoperational and operational cost components, this cost component is taken to be 20% of the total costs of all the above cost components.

C.5.2 Surveillance Costs

This section considers costs borne by the site operator during the surveillance period. This period occurs after the closure period and prior to the start of the institutional control period. During this period, the licensee would still be responsible for the care of the site and also be responsible for all site

Table C-25. Closure Period Equipment Unit Costs ($/machine-day)

Equipment	$/MD	Equipment	$/MD
Bulldozer	890	Water truck	200
Front end loader	420	Auger rig	600
Dump truck	215	Paving machine	355
Pan scraper	940	Tandem roller	170
Motor grader	420	Compactor	300
Backhoe	245	Farm tractor	50
40-ton crane	480	Hand tamper	12
100-ton crane	835	Cement truck	50
Forklift, small	140	Mobile cement pump with boom	460
Forklift, large	160		

maintenance and environmental monitoring activities. This responsibility would be maintained by the licensee until the license is transferred to the site owner.

It can be seen that at least three general considerations influence the level of costs during this the surveillance period:

- Waste volume (the size of the site);
- Site location (e.g., humid or arid); and
- Stability of disposed waste.

In regard to the last item, additional environmental monitoring costs would be expected for unstable waste disposal in addition to purely maintenance activities.

The principal cost components contributing to site surveillance costs are listed below:

Salaries
Administration
Regulatory
Environmental monitoring
Personal monitoring
Equipment costs
Other
Contingency

Each is briefly discussed below.

Salaries

Personnel contributing to this cost component essentially consists of one full-time site representative plus some part-time personnel. The full-time site representative is assumed to be approximately equivalent to a foreman. Annual salary and benefit costs therefore become:

Base Salary (x 1000):	$28
Benefit (30%):	8.4
Total:	$36.4

Part-time personnel are required for

- maintenance of structures and grounds;
- maintenance of disposal cells;
- radiation surveys;
- environmental monitorings; and
- quality assurance

Man-hours per appropriate individual are determined as a function of waste volume, site environment, and site stability. After increasing calculated man-hours by 75% to account for dead time, the total man-year requirements are determined per individual, assuming 220 working days per year. Fractions of a year are prorated to individual costs for an entire year to account for use of part-time personnel.

For maintenance and site surveys, the assumed personnel man-day requirements are listed below as a function of the total waste volume disposed at the site. Unit (man-day per m^3 of waste) requirements are assumed to be distributed among the listed crews.

Activity	Crew	Man-Day/(1000 m^3)
Maintenance of structures and grounds	1 QA technician 1 laborer 2 skilled laborers	0.1
Maintenance of disposal cells	1 QA technician 2 laborers 1 skilled laborer	unstable: 0.2(a)(b) stable: 0.005(a)(b)
Radiation surveys	1 QA technician 3 radiation safety technicians	0.05

[a] The first value is for waste disposal in a structurally unstable manner (either unstable or disposed with other unstable waste).

[b] These values are assumed to be halved for an arid disposal site.

Personnel duties also include collection of environmental samples, preparation of samples for shipment to an offsite laboratory for analysis, and interpretation of results. Personnel requirements are assumed to be a function of the number of samples analyzed per year. Personnel contributing to this cost factor include radiation safety technicians and quality assurance technicians as follows:

Personnel	Man-Day/Monitoring Cost ($)
Radiation safety technician	0.0075
QA technician	0.0025

Administration

During this time period a reduced level of activity takes place, with a corresponding reduction in administration costs. The largest reduction is believed to be associated with state or compact administration costs rather than company administration costs. Corporate administration costs are assumed to be mostly associated with the local company office located in the capital city of the state in which the facility is sited. This local office is assumed to be retained as source of site information until the license for the disposal facility is transferred to the site owner. This cost is therefore assumed to be $75,000 per year.

State or compact administration costs are assumed to be relatively reduced during this time period, since the principal focus of the compact during this time period would be directed toward selection of a new site. An annual cost of $75,000 is assumed.

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Regulatory

Regulatory costs are assumed to be reduced during this time period. Since no waste is being delivered for disposal, there is no need for an onsite inspector. However, the independent confirmatory monitoring program is continued. This is assumed to total $50,000 per year. An annual site inspection is also assumed at a cost of $20,000 per inspection. The total cost is therefore assumed to be $70,000/yr.

Environmental Monitoring

As for operational costs, this cost component covers sample analysis costs associated with isotope-specific radiation surveys and environmental monitoring, where samples obtained from the environmental monitoring program are sent offsite for radiochemical and gamma spectral analysis. The types of typical samples collected are listed in Table C-22.

As for operational costs, an environmental sampling cost is assumed which is a function of the volume of waste disposed, the site environment, and the stability of the disposed waste. This last factor is included given the expectation that unstable waste disposal implies a greater potential for environmental release, with corresponding additional sampling needs.

Environmental monitoring costs are thus given as follows:

Site Environment and Stability	Total Waste Volume Disposed (m^3)		
	50,000	500,000	1,000,000
Arid-unstable	41.5	75	110
Arid-stable	40.15	61.5	83
Humid-unstable	75	230	400
Humid-stable	61.5	95	130

Personnel Monitoring

This cost component is determined in the same manner as operational costs. Personnel man-days are determined per job occupation using the methodology discussed above under the "salaries" cost component. If a worker works only part of a year, the fraction is rounded up. The total number of workers that have spent at least some time on the site is determined and then multiplied by a yearly individual monitoring cost (assumed to be $600/yr/individual).

Equipment Costs

A certain amount of equipment is needed on site in order to perform maintenance and surveillance activities. One pickup truck is assumed to be maintained full time. Often pieces of equipment, assuming no major site problems, include a front end loader to transport fill and/or cobbles and a farm tractor (with attachments) for mowing and other activities. Machine-day requirements are given below as a function of the total volume of waste disposed at the site (in m^3).

Equipment	Machine-Days/(1000 m^3)	
Loader	Humid unstable:	0.25
	Humid stable:	0.06
	Arid unstable:	0.06
	Arid stable:	0
Farm tractor	Humid unstable:	0.25
	Humid stable:	0.25
	Arid unstable:	0
	Arid stable:	0

Yearly requirements are then determined by assuming 250 working days per year, plus an 80% efficiency factor.

Once the yearly fractional machine requirements are determined, costs are determined as a leasing charge plus an operating charge. Operating charges include fuel, oil, lubrication, normal expendables for the equipment, and a percentage of a mechanic's wages charged to maintenance. These costs are approximated based on weekly machine costs in reference 17 and are summarized below in units of \$/machine-day:

Vehicle	Leasing	Operating	Total
Pickup truck	30	40	70
Loader	280	140	420
Farm tractor	25	25	50

Other Costs

This cost component is composed of a number of assorted expenses, each of which would be expected to be relatively small. These expenses can include:

- Consulting and studies
- Legal fees
- Public outreach
- Insurance
- Utilities and supplies
- Communications
- Travel (e.g., to headquarters, scientific seminars, etc.)
- Temporary equipment rental
- Personnel management costs

Costs for these items totaled a constant \$680,000 during the closure period, but would be expected to be reduced during the surveillance period. A total \$300,000 is assumed.

Contingency

This cost component is estimated as 20% of all of the above cost components.

C.5.3 Institutional Control Costs

During this time period, the site is assumed to be under the control of a State agency. Surveillance and maintenance activities carried out by this agency are assumed to be monitored by the State health department or some other regulatory agency. Similarly to costs during the surveillance period, at least three general considerations tend to influence the level of costs during this time period. These are:

- Waste volume (the size of the site);
- Site location (e.g., humid or arid); and
- Stability of disposed waste.

The principal cost components contributing to institutional control costs are listed below:

Salaries
Administration
Regulatory
Environmental Monitoring
Personnel monitoring
Equipment costs
Other costs
Contingency

Each cost component is individually discussed below. It should be also noted that these costs correspond to the first few years of the institutional control period. Assuming that no problems are observed, it would be expected that reduced annual institutional control costs would be expected overtime. This is conservatively not considered, however, given the uncertainties in determining these costs as a function of time. Possible additional contingency costs may be added by the code user as discussed later.

Salaries

These costs are assumed to be the same as those during the surveillance period.

Administration

These costs are estimated somewhat differently than those during the closure or surveillance periods. Two contributors to these costs are assumed: those borne by the agency holding the site license, and a fraction of the compact administration costs. A annual cost of $50,000 is assumed for each contributor, or $100,000 per year.

Regulatory

These are assumed to mainly consist of occasional independent sampling and monitoring. A cost of about $30,000 is budgeted.

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Environmental and Personnel Monitoring

These are calculated in the same manner as similar costs during the surveillance period.

Equipment Costs

These are calculated in a similar manner as those during the surveillance period.

Other Costs

This cost component includes additional miscellaneous costs associated with site custody and maintenance. Typical types of expenses include:

- consulting and studies
- legal fees
- utilities and supplies
- communications
- travel (e.g., to and from the site)
- personnel management costs.

Based on similar types of expenses for the closure and surveillance periods, this cost component is estimated at about $200,000/yr.

Contingency

This cost component is estimated as 20% of all of the above cost components.

C.6 TIME VALUE OF MONEY

In this report a simple present value analysis is used to integrate the cost components discussed in the previous two sections, as well as other economic factors. A very simple description of the present value concept is given by the following equation (Ref. 15):

$$PV = FV/(1+d)^N \qquad (C-15)$$

where

PV = present value;
FV = future value;
d = rate of return or discount rate; and
N = number of time periods.

The rate of return is the discount rate that sets the net present value equal to zero. That is, the present values of cash outflows equals the present values of cash inflows.

Over the last few years, there have been published a number of present value analyses of economics involved with siting, operating, and closing a low-level waste disposal facility. Most analyses have involved shallow land burial facilities of various annual volumes and capacities. Some analyses have been of generic application while others have been oriented more toward specific sites. The analysis developed in this report is conceptual in nature (includes a number of simplifying assumptions), is principally oriented toward generic applications, and is also intended to be applicable to a number of various types of disposal technologies. Perhaps most importantly, the analysis methodology is meant to be used to compare disposal alternatives, and so the ability to compare costs is judged to be more important than the ability to perform a detailed cash flow analysis for a specific site and design. Development (and explanation) of the analysis methodology is very closely patterned after an approach taken by Baird and Rogers in "A Generalized Economic Model for Evaluating Disposal Costs at a Low-Level Waste Disposal Facility" (Ref. 35). This model was chosen since it gives a single cost number which can be used to compare alternatives. However, the analysis methodology also considers and draws upon analyses performed by other authors for waste disposal facilities and other applications (e.g., Refs. 12-15, 32).

In this simplified approach, the disposal facility is assumed to be sited and operated by a private corporation which operates the facility on a profit basis, and which also carries out other business activities in the general waste management field. The present values of the cash inflows and outflows are determined, and a unit disposal cost is determined so that the sum of the present values is set equal to zero. The lifetime over which the cash flows are discounted is the sum of the preoperational, operational, closure, and observation and surveillance periods. A unit institutional control surcharge is also determined which sets the present value of the institutional control costs equal to zero. All costs have been given in constant 1984 dollars, and so the models are formulated so that future costs are discounted to this year. Disposal facilities, however, may be initiated in alternative years, and so factors are included to enable adjustment of these costs.

C.6.1 Definitions

In this report, the costs associated with each of the five periods in the disposal facility lifetime have been broken down into cost components. Each cost component may be a function of the disposal facility size (e.g., waste volume) and/or disposal technologies used, or it may be constant. These cost components are summarized below.

Cost Components for Disposal Facility Lifetime

Preoperational	Operational	Closure	Surveillance	Inst. Control
Land	Salaries	Salaries	Salaries	Salaries
Licensing	Disposal cell materials	Equipment expenses	Environmental and personnel monitoring	Environmental monitoring
Administration	Environmental monitoring	Other costs	Equipment expenses	Personnel monitoring
Startup overhead	Personnel monitoring and training	Contingency	Other costs	Equipment costs
Heavy equipment	Heavy equipment operating expenses		Contingency	Other costs
Light equipment	QA and compliance testing			Contingency
Land development	Other costs			
Buildings	Contingency			
Utilities				
Engineering and design				
Contingency				

Cash flows throughout the life of the disposal facility are made up of various combinations of these cost components. Important cash flows are as follows:

POE(n) = Preoperational expenses: Cash flows during the preoperational period which are tax deductible as business expenses but not depreciable.

POC(n) = Preoperational costs: Cash flows during the preoperational period which are capitalized and thus depreciable but not tax deductible.

ITC = Investment tax credit: Tax credits which result from capital investment during the preoperational and operational periods. These tax credits are assumed to be available during the years that the investments are made.

R(n) = Revenues: Taxable revenues generated through waste disposal operations, and assumed to be a multiple of the annual volume of waste disposed. Revenues collected during the first year of disposal operations are designated R_1.

OE(n) = Operational expenses: Tax deductible expenses incurred during the operational period.

OC(n) = Operational costs: Costs during the operational period which are capitalized and thus depreciable but are not tax deductible.

D(n) = Depreciation expenses: Tax deductible allowance applied during the operational period for depreciation of capital costs.

SBC(n) = Surety bond premium (closure): Expenses occurred during the operational period for maintaining a surety bond sufficient to ensure closure funds.

SBI(n) = Surety bond premium (institutional control): Expenses occurred during the operational period for maintaining a surety bond sufficient to ensure institutional control funds.

CC(n) = Closure costs: Expenses occurred by the disposal facility operator during the closure period. No revenues are collected during this period.

SC(n) = Surveillance costs: Expenses incurred by the disposal facility operator during the observation and surveillance period. No revenues are collected during this period.

ICC(n) = Institutional control costs: Expenses incurred by the disposal facility site owner (not the operator) during the institutional control period. No revenues are assumed to be collected during this period.

ICS(n) = Institutional control surcharge: Tax exempt revenues collected by the disposal facility operator and placed in a sinking fund in order to pay for institutional control expenses. The revenues are collected as a surcharge on waste received at the disposal facility.

Other important parameters are as follows:

n = year during which expenses occur.

$V(n)$ = waste volume received annually during the operational period (m^3)

UC_1 = unit costs (\$/m^3) for disposal service during the first year of the operational period, and where $R_1 = UC_1 * V(n)$

USC = unit surcharge (\$/m^3) collected during the operational period for the institutional control fund, and where $ICS(n) = USC * V(n)$. USC is assumed to be a constant during the operational period.

UCA = average unit cost (\$/m^3) for disposal service over the operational period, as inflated.

$$= UC_1 * \sum_{n=1}^{IOPS} (1+f_o)^n/IOPS$$

f_p = rate of inflation during the preoperational period

f_o = rate of inflation during the operational period

f_c = rate of inflation during the closure period

f_s = rate of inflation during the observation and surveillance period

j = rate of inflation during the institutional control period

i = average interest rate of revenues placed in escrow for the institutional control fund

d = discount rate applied by the disposal facility operator

T = tax rate

sc = surety fraction for closure surety bond

si = surety fraction for institutional control surety bond

IPOS = lifetime of the preoperational period, assumed to be 5 years

IOPS = lifetime of the operational period (yrs)

ICLS = lifetime of the closure period (yrs)

IOBS = lifetime of the observation and surveillance period (yrs)

IINS = lifetime of the institutional control period (yrs)

C.6.2 Disposal Cost Calculations

The equations for each of the major cash flows are presented. This is followed by the equations for the unit disposal costs.

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Preoperational Expenses

Preoperational expenses are immediately tax deductible in the year that they occur. The present value (PV) of these expenses is as follows:

$$PV(POE) = (1-T) * \sum_{n=1}^{IPOS} (POE(n) * ((1+f_p)/(1+d))^n) \qquad (C\text{-}16)$$

Preoperational expenses are assumed to include the following:

Land
Licensing
Administration
Startup overhead
Contingency

Preoperational Costs

Preoperational costs are not deductible in the year that they are incurred, but must be capitalized. The present value of these costs is as follows:

$$PV(POC) = \sum_{n=1}^{IPOS} (POC(n) * ((1+f_p)/(1+d))^n) \qquad (C\text{-}17)$$

Preoperational costs are assumed to include the following:

Heavy equipment
Light equipment
Land development
Buildings
Utilities
Engineering and design
Contingency

Revenues

The annual revenues, R(n), are equal to the product of the unit costs, UC(n), and the annual waste volume, V(n). Unit costs are assumed to be a function of the operational year to account for inflation during this period. It can be expressed as follows:

$$UC(n) = UC_1 * (1+f_o)^n \qquad (C\text{-}18)$$

The present value of the revenues can therefore be expressed as:

$$PV(R) = UC_1 * (1-T) * ((1+f_p)/(1+d))^{IPOS} *$$
$$\sum_{n=1}^{IOPS} (V(n) * ((1+f_o)/(1+d))^n) \qquad (C\text{-}19)$$

Revenues are reduced by the tax burden.

Operational Expenses

These expenses occur during the operational period and are tax deductible. The tax obligation of the disposal facility operator is reduced by the factor T * OE(n), and so the present value of these expenses are as follows:

$$PV(OE) = (1-T) * (((1+f_p)/(1+d))^{IPOS}) * \sum_{n=1}^{IOPS} (OE(n) * ((1+f_o)/(1+d))^n) \qquad (C-20)$$

Operational expenses are assumed to include the following:

Salaries
Disposal cell materials
Environmental monitoring
Personnel training and monitoring
Heavy equipment operating expenses
QA and compliance testing
Other costs
Contingency

Operational Costs

These costs are comprised of costs for replacement of heavy and light equipment (including contingency) during the operational period. The present value of these costs is as follows:

$$PV(OC) = ((1+f_p)/(1+d))^{IPOS} * \sum_{n=1}^{IOPS} (OC(n) * ((1+f_o)/(1+d))^n) \qquad (C-21)$$

These costs are zero during most of the operational period, and are only expended at intervals associated with equipment lifetimes. An equivalent way to calculate these costs is to group the costs associated with each piece of equipment according to their lifetimes. The present value of costs for replacing any piece (or group) of equipment with life time, L, is thus:

$$PV(C_e) = ((1+f_p)/(1+d))^{IPOS} * \sum_{p=1}^{H} C_e * ((1+f_o)/(1+d))^I \qquad (C-22)$$

where C_e is the equipment cost (including contingency) in constant 1984 dollars, H is given as the integer truncation of the expression (IOPS-1)/L, and I = p*L.

Depreciation

Prior to the Economic Recovery Act of 1981, three of the most commonly used methods for calculating depreciation were the straight-line, declining balance, and sum-of-digits methods. Any reasonable method that was consistently applied, however, was also acceptable. Since the Act, both the terminology and the method of computing depreciation has changed. "Depreciable property" is now termed "recovery property", while "depreciation" is now termed "accelerated cost recovery system (ACRS) deduction." Depreciation for assets in service after 1980 is to be based either on ACRS percentage tables or on a straight-line method. Recovery periods are specified for various types of property (longer periods may also be used).

This study is of generic application, and so some simplifying assumptions are made. A straight-line approach is used since it is simple and is acceptable for assets placed in service both before and after passage of the Act. In actual application, a disposal facility operator would specify the most advantageous recovery periods for the various types of properties. In this study, depreciable items are assumed to consist of such assets as buildings, equipment, land development items (e.g., fences, roads), and utilities. Items such as buildings, land development items, and utilities are assumed to have a useful life equal to the life of the disposal facility. Heavy and light equipment are assumed to have useful lives equal to their replacement schedules. Depreciation is not considered during the preoperational period, and possible possible salvage values are assumed to be negligible.

The present value of depreciation for buildings, land development items, and similar assets is given as:

$$PV(D_b) = (T/(1+d)^{IPOS}) * \sum_{n=1}^{IOPS} (C_b/IOPS) * (1/(1+d))^n \qquad (C\text{-}23)$$

where C_b = building and other costs

$$= \sum_{m=1}^{IPOS} BC(m) * (1+f_p)^m$$

BC(m) = sum of annual preoperational cost components for buildings, land development items, utilities, and engineering and design (includes contingency).

Depreciation for heavy and light equipment is more complicated since the lifetimes are normally less that the length of the operational period. Assuming that a piece of equipment has a lifetime, L, then the depreciation over the first L years of the facility operational life is:

$$D_e(n) = (1+f_p)^{IPOS} * (C_e/L) \quad (n = 1, L) \qquad (C\text{-}24a)$$

G/a

Similarly,

$$D_e(n) = (1+f_p)^{IPOS} * (C_e/L) * (1+f_o)^{L} \qquad (n = L+1,\ 2L) \qquad \text{(C-24b)}$$

$$D_e(n) = (1+f_p)^{IPOS} * (C_e/L) * (1+f_o)^{2L} \qquad (n = 2L+1,\ 3L) \qquad \text{(C-24c)}$$

and so forth until n equals IOPS, where C_e is the equipment cost (including contingency) in constant 1984 dollars.

The present value of the equipment depreciation can then be expressed as:

$$PV(D_e) = T * ((1+f_p)/(1+d))^{IPOS} * \sum_{p=1}^{H} ((1+f_o)/(1+d))^{I} * \sum_{n=1}^{L} C_e/(L * (1+d)^{n}) \qquad \text{(C-25)}$$

where H is given as the integer truncation of the expression ((IOPS-1)/L) + 1, and I = (p-1)*L.

Investment Tax Credit

The investment credit is applied to the acquisition of equipment, buildings, and other items pursuant to the regulations of the U. S. Internal Revenue Service. From 1975 to 1983, the tax credit was based on 10% of the investment. Since the beginning of 1983, modified tax regulations allow two alternative ways to calculate and apply the credit. One approach is to apply 8% of the qualified investment plus assume full value for depreciable items. Another approach is to apply 10% of the qualified investment plus reduce the depreciable basis of the property by 50% of the investment credit claimed. (Given the generic nature of this study, the first, simpler, approach is taken.) By whichever approach, the "qualified investment" depends upon the recovery period of the depreciable property: 60% is applied to 3-year property; and 100% is applied to 5-, 10-, and 15-year property. The tax credit is furthermore limited to $25,000 plus 85% of any tax liability over $25,000. Any unused tax credit may be carried over to the 3 preceding tax years plus the 15 succeeding tax years.

Calculation of the investment credit can become quite complicated. The most significant complication is that a ceiling is imposed on the amount of credit that can be annually applied, and that this ceiling is calculated based on the tax liability. To determine the tax liability, one must know the annual revenues, operational expenses, depreciation, and so forth. Since annual revenues are determined as the product of the unit charge and the annual waste volume, and the unit charge is precisely the unknown variable to be determined in this chapter, calculation of the annual tax liability requires an iterative approach.

Given the generic nature of this study, a simplifying assumption is made that all of the eligible investment credit determined for any given year is applied during that year. Investment credits are thus modeled as "spikes" throughout the preoperational and operational periods, while in reality distribution of the credits would probably be smoothed out -- i. e., credits would actually be applied at reduced levels both preceding and following the spikes.

With this assumption, the present value of the investment credit can be determined for both preoperational (ICP) and operational (ICO) costs as follows:

$$PV(ITC) = PV(ICP) + PV(ICO) \qquad \text{(C-26)}$$

The present values of the investment credit associated with preoperational costs is approximated as:

$$PV(ICP) = 0.08 * \sum_{n=1}^{IPOS} POC(n) * ((1+f_p)/(1+d))^n \qquad \text{(C-27)}$$

where preoperational costs are given as:

- Heavy equipment
- Light equipment
- Land development
- Buildings
- Utilities
- Engineering and design
- Contingency

Investment credits for the operational period are associated with acquisition of replacement heavy and light equipment, and therefore occur at intervals throughout the operational life. Similar to the above operational cost calculations, the present value of the investment credit for any piece (or group) of equipment with lifetime, L, is:

$$PV(ICO) = 0.08 * ((1+f_p)/(1+d))^{IPOS} * \sum_{p=1}^{H} C_e * ((1+f_o)/(1+d))^I \qquad \text{(C-28)}$$

where C_e is the equipment cost in constant 1984 dollars, H is given as the integer truncation of the expression (IOPS-1)/L, and I = p*L.

Closure Costs

These costs are expended by the disposal facility operator during the closure period. They are not capitalized but are assumed to be tax deductible. This is because closure activities are required by regulation, and therefore closure costs must be expended in order to operate the disposal business. The present value of these costs is as follows:

$$PV(CC) = (1-T) * ((1+f_p)/(1+d))^{IPOS} * ((1+f_o)/(1+d))^{IOPS} *$$

$$\sum_{n=1}^{ICLS} (CC(n) * ((1+f_c)/(1+d))^n) \qquad \text{(C-29)}$$

In this report, closure is normally assumed to require only one year, and costs are calculated automatically based on the assumptions presented in Section C.5.1. The code user, however, has two ways in which to modify these assumptions. First, the code user may input an alternative number of closure years, in which case the closure costs calculated in Section C.5.1 will be applied to each of the closure years. Second, the code user may optionally override the Section C.5.1 calculations and input closure costs, in constant 1984 dollars, for as many years as specified during the closure period (i.e., if three years are specified, 3 costs values must be input). The costs are automatically inflated as provided above. This option allows the code user to consider additional contingencies.

Surveillance Costs

These costs are incurred by the disposal facility operator after closure activities and prior to transfer of the disposal facility license to the site owner. The present value of these costs, which like closure costs are not capitalized but are assumed to be required by regulation and are thus a deductible business expense, are as follows:

$$\mathrm{PV(SC)} = (1-T) * ((1+f_p)/(1+d))^{\mathrm{IPOS}} * ((1+f_o)/(1+d))^{\mathrm{IOPS}} * ((1+f_c)/(1+d))^{\mathrm{ICLS}} * \sum_{n=1}^{\mathrm{IOBS}} (\mathrm{SC}(n) * ((1+f_s)/(1+d))^n) \qquad \text{(C-30)}$$

The code user again has two options for performing the calculations. First, IOBS is specified by the code user. If only IOBS is specified, surveillance costs calculated using the techniques presented in Section C.5.2 are applied during each of the surveillance period years. Second, the costs calculated in Section C.5.2 may be replaced at the code user's option in an identical manner as that provided above for closure costs.

Surety Bond Premium (Closure)

These are business expenses occurred during the operational period to ensure that sufficient money will be available to close the disposal facility. The surety bond costs are assumed to be tax deductible since they are assumed to be required, either by regulation or otherwise as a condition of the land owner, in order to operate the disposal facility.

The approach taken is to determine for each operational year the total costs that would be required to close the disposal facility assuming that closure was required at the end of that operational year. The annual surety bond premium, SBC(n), is then assumed to be a fraction, sc, of these annual expenses. This is consistent with the intent of the surety bond -- i.e., to ensure that funds will be available if for some reason closure is required at any time over the operating life of the facility. A simplifying assumption is made in that base (uninflated) closure costs are used which are calculated for a facility at the end of the operational life. Barring an unexpected contingency, one would expect that closure would actually be somewhat less if it took place earlier during the operational life (some of the closure costs components are waste volume dependent).

The surety bond premium required during each year (n) of the operational period, as inflated from the beginning of the preoperational period, is given as follows:

$$SBC(n) = sc * (1+f_p)^{IPOS} * (1+f_o)^n * \sum_{m=1}^{ICLS} CC(m) * (1+f_o)^m \qquad (C\text{-}31)$$

where m signifies the particular year considered in the closure period.

The present value of these premiums is thus given as:

$$PV(SBC) = (1-T) * sc * ((1+f_p)/(1+d))^{IPOS} *$$

$$\sum_{n=1}^{IOPS} [((1+f_o)/(1+d))^n * \sum_{m=1}^{ICLS} (CC(m) * (1+f_o)^m)] \qquad (C\text{-}32)$$

Surety Bond Premium (Institutional Control)

Surety bonding may also be required of the disposal facility operator in order to ensure that sufficient funding is available to the site owner to carry out maintenance activities during the institutional control period. Over the course of the operational period, money is collected from disposal facility customers and placed in a sinking fund, and so whatever funds that had been accumulated during the operational period up to the time considered would be available to the site owner. The facility operator is thus assumed to only require a surety sufficient to fund the difference between the total cost of the institutional control period and the amount of money accumulated in the sinking fund. The annual surety charge during the operational period is thus given as:

$$SBI(n) = ss * (1+f_p)^{IPOS} * (1+f_o)^n *$$

$$[\sum_{m=1}^{IINS} A * G - \sum_{p=1}^{n} V(p) * (1+i)^{ICLS} * USC * (1+i)^p] \qquad (C\text{-}33)$$

where A is assumed to be $(1+f_o)^{ICLS}$, and G is defined in Section C.6.3 (Institutional control costs). The present value of these expenses is as follows:

$$PV(SBI) = (1-T) * ss * ((1+f_p)/(1+d))^{IPOS} * \sum_{n=1}^{IOPS} ((1+f_o)/(1+d))^n *$$

$$[\sum_{m=1}^{IINS} A * G - \sum_{p=1}^{n} V(p) * (1+i)^{ICLS} * USC * (1+i)^p] \qquad (C\text{-}34)$$

Unit Disposal Cost

Unit disposal costs are determined by setting the sum of the present values of the cash inflows and outflows equal to zero. That is:

$$0 = PV(ITC) + PV(R) - PV(POE) - PV(POC) - PV(OE) - PV(OC) + PV(D) - PV(CC) - PV(SC) - PV(SBC) - PV(SBI) \qquad \text{(C-35)}$$

Solving for the unit disposal costs, UC_1, one has:

$$UC_1 = A/B \qquad \text{(C-36)}$$

where,

$$A = PV(POE) + PV(POC) + PV(OE) + PV(OC) - PV(D) + PV(CC) + PV(SC) + PV(SBC) + PV(SBI) - PV(ITC)$$

$$B = (1-T) * ((1+f_p)/(1+d))^{IPOS} * \sum_{n=1}^{IOPS} (V(n) * ((1+f_o)/(1+d))^n)$$

As discussed above, the unit cost for any operational year is assumed to be:

$$UC(n) = UC_1 \; (1+f_o)^n \qquad \text{(C-19)}$$

The average unit cost over the operational period is:

$$UCA = UC_1 * \sum_{n=1}^{IOPS} (1+f_o)^n/IOPS \qquad \text{(C-37)}$$

C.6.3 Institutional Control Costs

The methodology for calculating institutional control costs is similar to that used for the final Part 61 EIS (Ref. 2), except that much more flexibility has been adopted to increase the general utility of the methodology to the user. Institutional control costs are calculated assuming that money is collected as a surcharge on waste received at the facility and deposited in a sinking fund. The object of the calculational methodology is to determine the required magnitude of the surcharge as a function of costs expended over the institutional control period and various economic parameters such as interest and inflation rates.

The remainder of this section addresses present value calculations for institutional control costs and the institutional control cost surcharge. This is followed by the equation for determining the surcharge.

Institutional Control Costs

The present value of the institutional control costs is given by:

$$PV(ICC) = ((1+f_p)/(1+i))^{IPOS} * ((1+f_o)/(1+i))^{IOPS} * ((1+f_c)/(1+i))^{ICLS} * ((1+f_s)/(1+i))^{IOBS} * G \qquad \text{(C-38)}$$

where i represents the interest rate provided by a very secure investment. The factor G is given as follows:

$$G = C_{t1} * \sum_{n=1}^{a} ((1+j)/(1+i))^{n} + C_{t2} * \sum_{n=a+1}^{b} ((1+j)/(1+i))^{n} +$$

$$C_{t3} * \sum_{n=b+1}^{c} ((1+j)/(1+i))^{n} + . . . + C_{tm} * \sum_{n=k+1}^{IINS} ((1+j)/(1+i))^{n} \quad \text{(C-39)}$$

where C_{t1}, C_{t2}, C_{t3}, . . ., are costs for various levels of institutional control activities. (C_{t1}, C_{t2}, C_{t3}, . . ., are all in 1984 dollars.) The subscripts t1, t2, t3, . . ., refer to the time intervals over which the costs must be expended, where the total number of time intervals is m. Obviously, m must be less than or equal to IINS (t1 + t2 + t3 + + . . . + tm = IINS). Other parameters are as follows:

n = any year during the institutional control period

j = average inflation rate over the institutional control period

i = average interest rate of the sinking fund

a = t1

b = t1 + t2

c = t1 + t2 + t3

k = t1 + t2 + t3 + t_{m-1}

The code user has three options for proceeding with the calculations. First, the code user may merely input IINS, in which case the institutional control costs determined in Section C.5.3 are applied for each year of the institutional control period ($C_{t1} = C_{t2} = C_{t3}$, etc.). Second, the code user may optionally bypass the Section C.5.3 calculations and substitute alternative institutional control costs. One merely inputs the costs in constant 1984 dollars, plus the lengths of time that the costs are applicable. Third, the code user may optionally add lump-sum costs to specific years of the institutional control period.

Institutional Control Surcharge

Funding for institutional control is provided by a surcharge on waste received at the disposal facility. Funds are placed in a sinking fund by the site operator. The present value of these costs is given as:

$$PV(ICS) = USC * (1/(1+i))^{IPOS} * \sum_{n=1}^{IOPS} V(n) * (1/(1+i))^{n} \quad \text{(C-40)}$$

USC is assumed to be constant over the operating life of the disposal facility; hence, inflation is not considered in this equation.

Surcharge Unit Cost

The unit cost for the institutional control surcharge is calculated by setting the sum of the present values equal to zero. That is:

$$PV(ICS) - PV(ICC) = 0 \qquad (C\text{-}41)$$

Therefore,

$$USC = PV(ICC)/[(1/(1+i))^{IOPS} * \sum_{n=1}^{IOPS} V(n) * (1/(1+i))^{n}] \qquad (C\text{-}42)$$

The total cost charged to the disposal facility customer is thus equal to UC(n) + USC.

C.6 4 Additional Considerations

Two additional considerations relate to the disposal rate structure and the discount year.

Disposal Rate Structure

Equation C-37 calculates an average unit price over all waste delivered to the disposal facility, and also over all disposal technologies used. It contains a number of simplifying assumptions -- for example, there is no provision to track debt and equity capital separately -- although it is believed to be generally a reasonable approximation since the analysis methodology is designed for generic, conceptual applications rather than site-specific applications.

One large limitation is the inability to estimate the overall rate structure that would be applied at the disposal facility. That is, it is possible to consider six different disposal technologies at the disposal site, and it is highly unlikely that a disposal facility operator would charge the same price for shallow land burial, for example, as for a concrete bunker. The unit cost charged for the bunker would undoubtedly exceed that charged for shallow land burial. Another factor is the radiation level of the waste packages. Since handling is more difficult for waste packages having elevated radiation levels, current disposal facility operators both apply graduated surcharges based on radiation level. Both current disposal facility operators also apply graduated surcharges based on the total activity content of a waste package.

Discount Year

The various cost components for the five periods in the life of the disposal facility have all been given in 1984 dollars, and the equations presented in the previous sections of this chapter have been formulated assuming that the present values are discounted to this year. That is, 1984 is assumed to be the year that preoperational activities begin.

The code user has the option to commence preoperational activities to alternative years -- i.e., any year between 1985 and 1990. Discounting to 1975 is allowed in order to provide a mechanism to roughly compare the results of the calculations from this report with those in the final Part 61 EIS (Ref. 2). A 1990 cap is assumed since the waste volume projections extend to the year 2030 in this report, and this ceiling would allow for a relatively large disposal facility lifetime (35 years operational life after a 5-year preoperational period).

Thus, the code user inputs the selected discount year plus the desired adjustment (e.g., inflation) rate. The cost components are then inflated (or deflated) as given by the following factor, F:

$$F = C * (1+h)^m \qquad \text{(C-43)}$$

where,

C = base (1984) dollars

h = selected adjustment factor

$m = y - 1984$

y = selected discount year

C.7 REFERENCE SITES AND ENVIRONMENTAL CHARACTERISTICS

Calculation of radiological impacts from processing and disposal of waste is influenced by the environmental characteristics at the waste treatment and disposal location. It is furthermore expected that the user of the impact analysis codes will be mostly interested in generic applications rather than site-specific applications. Given the above, two approaches have been adopted to include environmental considerations into the calculational methodology. First, a set of four different reference treatment/disposal site environmental characteristics is assumed, and any of the four may be specified by the user of the codes. The assumed parameters specific to these site environments may be modified by the code user or different parameter values substituted. Second, all site environmental characteristics (or calculational factors which are derived from specific site environmental characteristics) are stored in a separate input file in the computer codes, and the user may optionally expand the file to include additional reference sites.

The environmental characteristics for the reference treatment/disposal sites are derived from the four regional disposal sites assumed for the environmental impact statements for the regulation 10 CFR Part 61 on low-level radioactive waste disposal (Refs. 1, 2). Some slight differences from the Part 61 regional sites are assumed, however, particularly regarding parameters relating to groundwater impacts. These reference sites include what has been termed the northeast site, the southeast site, the midwest site, and the southwest site. Site characteristics for any one of the reference sites are called by the user by inputting the appropriate integer value for the region index, IR (see Chapter 2.0). A brief descriptive summary of the environmental characteristics of each of the reference sites follows:

IR	Site	Precipitation	Soil Characteristics	Population Density
1	northeast (NE)	humid	low permeability	high
2	southeast (SE)	humid	moderate permeability	moderate
3	midwest (MW)	humid	low permeability	low
4	southwest (SW)	semi-arid	high permeability	low

The remainder of this chapter is separated into two sections. Section C.7.1 summarizes the basic environmental properties assumed for each reference site. Section C.7.2 presents the parameters which are used in the computer codes and are derived from the site-specific environmental parameters.

C.7.1 Environmental Properties

The basic environmental properties assumed for the reference sites have been developed from a number of sources and are meant to be typical of environmental conditions within a particular area of the country. The assumed environmental parameters, however, should not be interpreted as representing any existing disposal site or specific location.

Most of the base environmental properties associated with each reference site are summarized in Table C-26. As shown, the first three sites are located in humid environments while the last site is located in a semi-arid environment.

The average annual natural percolation (PERC) into the groundwater system refers to the amount of rainfall falling onto a site that actually infiltrates into the groundwater system beneath the site. It has been calculated for each reference site using actual representative data (Ref. 33). The calculation involves a mass balance of precipitating water taking into account factors such as infiltration, runoff, evaporation, and transpiration. For the southwest site, a water balance calculation indicates that only a negligible quantity of precipitating water actually enters the groundwater system. For conservativeness, however, and to provide a driving force for the calculations, the percolation for this site is assumed to be 1 mm per year.

The PE index is the Thornthwaite precipitation-evaporation index that is indicative of the antecedent moisture conditions of the soil and is commonly used to differentiate between the dusting potential of soils in different climatic divisions. The PE index is presented in Figure C.16 for the conterminous 48 states of the union (Ref. 33). The PE index and the average silt

Table C-26. Summary of Reference Treatment and Disposal Facility Site Environmental Properties

	Reference Sites			
Environmental property	NE(1)*	SE(2)*	MW(3)*	SW(4)*
Mean average temperature °C (°F)	8°C (46°F)	17°C (63°F)	11°C (51°F)	14°C (57°F)
Average wind speed m/sec (mph)	4.61 (10.3)	3.61 (8.1)	4.72 (10.6)	6.67 (15.5)
No. of days per year having at least 0.01 in precipitation	146	115	110	65
Average annual precipitation mm (in)	1,034 (41)	1,168 (46)	777 (30.5)	485 (19)
Average annual natural percolation (PERC) into groundwater system mm (in)	75 (2.9)	180 (7.1)	50 (2.0)	1 (.04)
Precipitation-evaporation (PE) index of site vicinity	136	91	93	21
Average silt content of site soils (%)	65	50	85	65
Average cation exchange capacity (meq/100g)	15	10	12	5
Travel time (yrs), waste to water table	0	10	70	276
Groundwater speed (m/yr)	0.1	1.25	0.66	10

*The numbers in parentheses denote values for the region index, IR, which are used in the codes to specify environmental parameters for each of the reference sites (e.g., if IR=1, then environmental parameters assumed for the northeast site are used in the calculations).

650

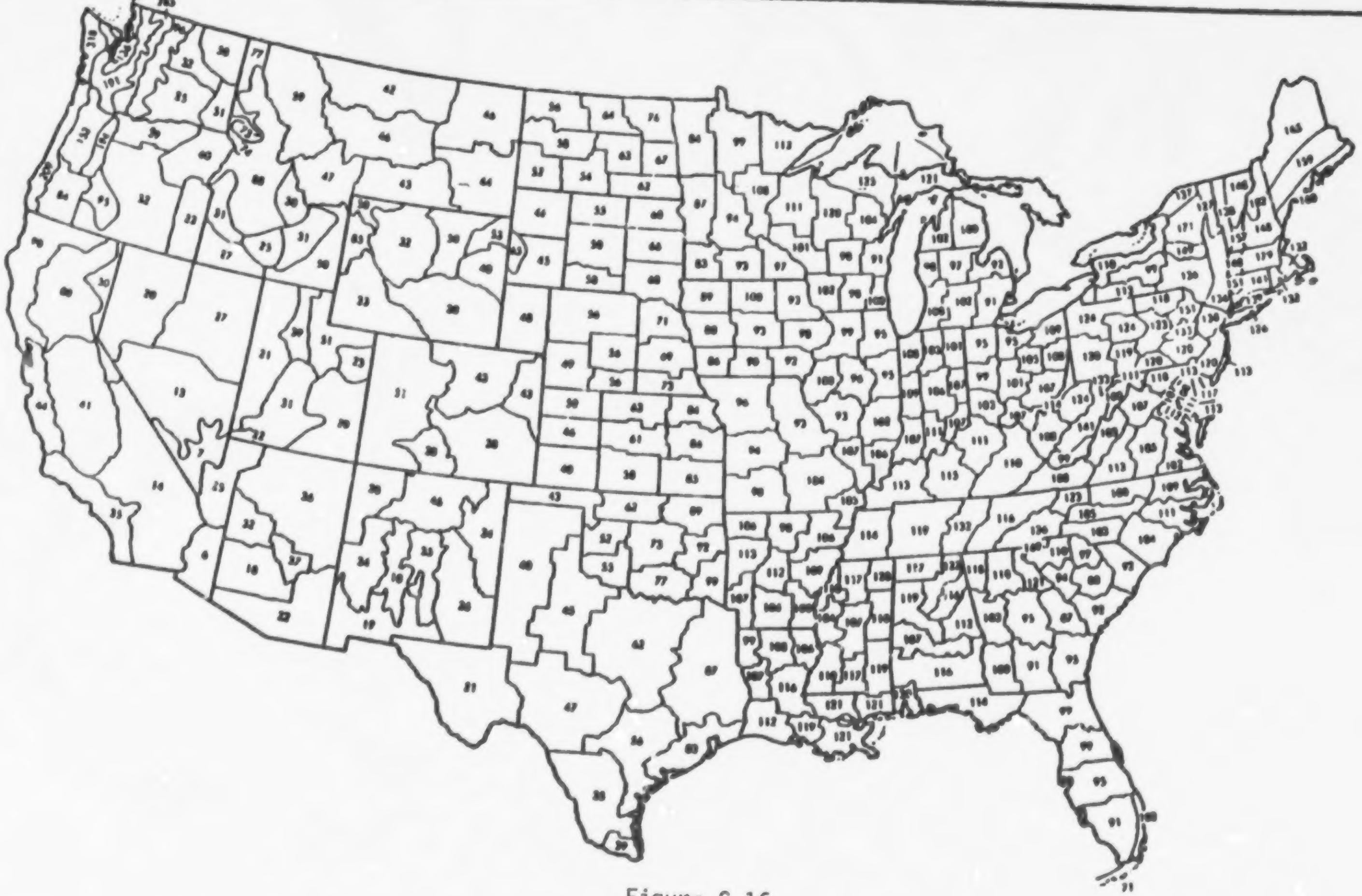

Figure C.16

MAP OF THORNTHWAITE'S PRECIPITATION-EVAPORATION INDEX VALUES FOR STATE CLIMATIC DIVISIONS content of the site soil are used to calculate impacts due to potential inadvertent intrusion into locations of disposed waste.

The average cation exchange capacity of the site soils is used as a guide for selecting retardation coefficients for purposes of calculating impacts from groundwater migration of radionuclides from disposal locations. The groundwater travel time and groundwater speed are also used in the groundwater migration calculations.

The base population distributions for each of the reference sites are summarized in Table C-27. These population sets are used to determine impacts from airborne releases of radionuclides from facilities located in "rural" environments. For facilities located in "urban" environments, population distributions are assumed which are a factor of 10 higher. Other population distributions are assumed for impacts calculated several years following disposal facility closure -- e.g., from erosion of a disposal facility several hundred to a few thousand years following disposal. These assumptions are addressed below.

C.7.2 Derived Parameters

Based on the above, a number of derived parameters may be obtained which are used in the calculations. Some of these derived environmental parameters are independent of the particular treatment/disposal method assumed, and are selected by the code user by inputting the desired value of the region index (IR). Other derived parameters are less straightforward, however, since these latter parameters are a function of the assumed treatment or disposal method in addition to the site environment. Reference facilities are assumed in this report, and so these dependent parameters are derived in this report and presented in this appendix. Options are also incorporated for use of alternative dependent environmental parameters supplied by the user of the code.

In the remainder of this section, the independent derived environmental parameters are first summarized, followed by the dependent derived environmental parameters. Finally, user supplied and override environmental parameters are summarized. Parameters used in the calculations which are totally independent of the site environment are discussed in the main body of this report.

Table C-27. Base Population Distributions for Reference Sites

Distance from Facility	Northeast	Southeast	Midwest	Southwest
0-5 miles	3,440	2,024	3,070	59
5-10 miles	20,513	8,115	4,998	180
10-20 miles	73,636	36,000	27,890	3,529
20-30 miles	121,559	124,995	104,181	9,062
30-40 miles	556,639	203,435	121,893	4,888
40-50 miles	1,012,788	104,933	359,146	27,158

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C.7.2.1 Independent Derived Environmental Parameters

The derived environmental parameters which are used in this report and are independent of the particular method used to treat or dispose of the waste are summarized in Table C-28. The terminology "symbol" refers to the variable name used for the particular parameter in the computer codes.

Corresponding environment-specific values for each of these parameters are given in Table C-29. FSC stands for the site-selection factor (f_s) used for determining airborne impacts stemming from the intruder-construction scenario at a closed disposal site. It is a function of the personnel exposure time, the wind speed at the site, the silt content of the site soil, and the precipitation-evaporation index of the site vicinity. Its derivation is discussed in Section 4.2.2.1 of the main body of this report. Similarly, FSA is the site selection factor used for determining airborne impacts from the intruder-agriculture scenario at a closed disposal site. Its derivation is discussed in Section 4.2.4.1.

The parameters QFC and NRET are used as part of determining impacts from groundwater migration. QFC is the dilution factor (Q) and is discussed in Section 4.3.2.2. It is the minimum volume of water (in m^3) into which radionuclides released into ground or surface water are diluted. NRET is a dimensionless integer index used to select a particular set of retardation coefficients (RET) for use in the calculations. (See Section 4.3.2.3.)

Any one of five possible sets of retardation coefficients may be selected by the user of the code (see Table 4-11 in Section 4.3.2.3); for this report, however, each site environment is assigned a particular NRET set.

The parameters UVEL (m/yr), UTCK (m), SVEL (m/yr), DSUR (m), and DISP (m) are also used in groundwater migration calculations. UVEL is the speed of water movement through the unsaturated zone, while UTCK is the thickness of the unsaturated zone. These parameters are discussed in Section 4.3.2.3. SVEL, the speed of water travel through the saturated zone, is discussed in Section 4.3.2.4 (also see Table 4-10).

The distance from the center of the disposal facility to the nearest surface water body is denoted by DSUR. Three other biota access locations are also considered for groundwater migration: a population well located half the distance between the disposal facility and the water body (DSUR/2), a boundary well located on the disposal site boundary, and an intruder well located at the downgradient edge of the disposal waste. Calculation of distances from the waste to the boundary and intruder wells, plus other groundwater parameters, is discussed in Section C.7.2.2. DISP is the dispersivity for water movement in the saturated zone.

The parameter POP is used in several scenarios in this report involving impacts to populations due to airborne releases. These scenarios include airborne releases from: (1) waste incineration (Section 3.1.1), (2) intruder-initiated exposed waste scenarios (Section 4.5.1), and (3) operation of a leachate evaporator (Section 4.4). Basically, POP is the population-weighted sum of the atmospheric dispersion factor (X/Q) as a function of the radial distance from a release source (units of person-yr/m^3). POPE is very similar to POP and is used to estimate impacts to populations due to airborne releases associated

Table C-28. Definitions of Independent Derived Environmental Parameters

Symbol	Scenario	Environmental Property
FSC	Intruder-construction	Site selection factor for airborne release
FSA	Intruder-agriculture	Site selection factor for airborne release
QFC	Groundwater migration	Dilution factor (m^3)
NRET	Groundwater migration	Retardation coefficient set
UVEL	Groundwater migration	Unsaturated zone water speed (m/yr)
UTCK	Groundwater migration	Unsaturated zone thickness (m)
SVEL	Groundwater migration	Saturated zone water speed (m/yr)
DSUR	Groundwater migration	Distance from the center of the disposal facility to the nearest downgradient surface water location (m)
DISP	Groundwater migration	Dispersivity of the saturated zone (m)
POP	Several scenarios	Transfer factor for airborne impacts to offsite populations (people-yr/m^3)
POPE	Exposed waste scenario	Transfer factor for airborne impacts from erosion (people-yr/m^3)
POPW	Exposed waste scenario	Transfer factor for waterborne impacts (people-yr-yr/m^3)
EERO	Exposed waste scenario	Airborne mobilization rate due to erosion (g/m^2-sec)
EINT	Exposed waste scenario	Airborne mobilization rate due to intruder (g/m^3-sec)
TPO	Transportation	Population density along transport route (people/$mile^2$)
TDP	Transportation	Calculational factor TDOZ for population impacts $(miles/ft)^2$
TDO	Transportation	Calculational factor TDOZ for occupational impacts $(miles/ft)^2$
VEL	Transportation	Transport vehicle speed (miles/hr)
WVEL	Several scenarios	Site wind speed (m/sec)
AXOQ	Accident-dropped container	X/Q factor for single container accident (yr/m^3)
FXOQ	Accident-fire	X/Q factor for accidental fire (yr/m^3)
RI	Pathway dose conversion factors	Irrigation rate of garden crops (m^3/m^2-day)
F3	Pathway dose conversion factors	Consumption of plants by animals (kg/day)

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Table C-29. Independent Derived Environmental Parameters

Symbol	Scenario/Env. Prop.	Reference Site Environments NE	 SE	 MW	 SW
FSC	Intruder-construction	9.18E-12	2.01E-11	2.51E-11	2.64E-10
FSA	Intruder-agriculture	2.96E-11	3.18E-11	3.28E-11	8.06E-11
	Dilution factors for groundwater migration				
QFC(1)	· Intruder well	7.7E+3	7.7E+3	7.3E+3	7.7E+3
QFC(2)	· Boundary well	7.7E+3	7.7E+3	7.7E+3	7.7E+3
QFC(3)	· Population well	2.0E+5	2.0E+5	2.0E+5	2.0E+5
QFC(4)	· Surface water	4.5E+6	4.5E+6	4.5E+6	NA
NRET	Retardation coefficient set used for groundwater migration	4	3	3	2
UVEL	Unsat. zone speed	0.1	0.2	0.1	0.1
UTCK	Unsat. zone thickness	0	2	7	28.
SVEL	Sat. zone speed	0.1	1.25	0.66	10.
DSUR	Distance to surface water	1000.	1000.	2000.	NA*
DISP	Sat. zone dispersivity	0.05	0.05	0.05	0.05
	Airborne transfer factors for population impacts				
POP**	· Intruder exposed waste	1.01E-9	3.50E-10	3.86E-10	2.66E-11
POPE*	· Erosion exposed waste	1.51E-9	5.25E-10	5.79E-10	3.99E-11
POPW	Waterborne transfer factor for exposed waste	1.11E-7	1.11E-7	1.11E-7	NA
EERO	Airborne mobilization rate, erosion	5.53E-7	1.54E-8	2.05E-6	7.95E-6
EINT	Airborne mobilization rate, intruder	2.03E-6	2.50E-6	4.49E-6	6.84E-5
TPO	Transportation population density	2280	610	790	60
TDP	Transportation factor, population	7.06E-5	7.06E-5	7.06E-5	3.92E-5
TDO	Transportation factor, occupational	9.57E-7	9.57E-7	9.57E-7	9.57E-7
VEL	Transport vehicle speed	50.	50.	50.	50.
WVEL	Site average wind speed	4.61	3.61	4.72	6.67
AXOQ	Waste accident X/Q factor	9.68E-11	1.40E-10	6.21E-11	4.11E-11
FXOQ	Fire accident X/Q factor	1.83E-9	1.83E-9	1.83E-9	1.83E-9
RI	Irrigation rate for PDCF	6.80E-4	6.80E-4	1.10E-3	2.70E-3
F3	Plant consumption for PDCF	25.	36.	22.	36.

*The nearest downgradient surface water body flows only intermittently. A DSUR distance of 5000 m is assumed, however, in order to enable calculation of the population well distance for the southwest site.

**Values given for rural environment (IPOP=2). For an urban environment (IPOP=1), multiply by 10.

with the erosion-initiated exposed waste scenario. (See Section 4.5.1.) It is determined by multiplying POP by a factor of 3 to account for population growth in the site area during the period prior to initiation of the scenario.

In this report, reference values for POP and POPE are assumed for each reference site. An example derivation of POP for a particular site is given below.

As discussed in Section 3.1.1, X/Q for a site atmospheric condition consisting of 1/3 stability Class C, 1/3 stability Class D, and 1/3 stability Class F may be expressed as a ground level release as follows:

$$X/Q = 4.168 \times 10^{-8}\ q(r)/r^2 \qquad \text{(C-44)}$$

where X/Q is in units of yr/m^3, r is the distance from the source point in meters, and q(r) is given as:

$$q(r) = 0.133\ (1 + .0002r)^{\frac{1}{2}} + 0.178\ (1 + .0015r)^{\frac{1}{2}} + 1 + .0003r \qquad \text{(C-45)}$$

For each site environment, POP is calculated as follows: obtain X/Q from the above equations for the midpoint of the distance from the source point, multiply X/Q by the population in the sector slice, and sum over all distances. As an example for the southeast site (see Table C-27):

r (miles)	r (meters)	X/Q	Population	POP
2.5	4,023	7.36E-15	2,024	1.49E-11
7.5	12,070	1.62E-15	8,115	1.31E-11
15	24,140	6.90E-16	36,000	2.48E-11
25	40,233	3.83E-16	124,995	4.78E-11
35	96,326	2.63E-16	203,435	5.35E-11
45	72,419	2.00E-16	104,933	2.09E-11
			Total:	1.75E-10

These reference values of POP and POPE are assumed to be applicable to a rural environment. For operations taking place in an urban environment, the calculated reference values are multiplied by 10. (For POP and POPE calculations involving user-supplied override parameters, no such multiplication by 10 is performed.)

POPW is discussed in Section 4.5.2 and is used as the site selection factor (f_s) for the waterborne releases associated with the intruder- and erosion-initiated exposed waste scenarios.

EERO and EINT denote the airborne mobilization rates, respectively, for the erosion-initiated and intruder-initiated exposed waste scenarios. These mobilization rates are in units of g/m^2-sec and are discussed in Section 4.5.1. EERO represents dust mobilization due to ambient wind conditions.

TPO is the population density (persons/mile^2) along the transport route. It is apparent that the population density can be highly variable, depending upon the region of the country considered and the relationship of waste generators,

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transport routes, and waste treatment/disposal facilities to population centers. Waste delivered to a treatment or disposal facility, for example, may originate in an area of high population density and be treated/disposed in an area of low population density.

Given the generic nature of this study, it would be counterproductive, not to mention impossible, to attempt to provide a detailed population density distribution for all possible transportation routes between all possible waste generators and all possible waste treatment/disposal facilities. What is done, rather, is to estimate average population densities in the vicinity of each of the reference site environments considered in this report. As discussed earlier, these reference site environments are derived from the regional disposal facility sites considered for the Part 61 EIS (Ref. 1). These regional sites were all assumed to be located in rural environments.

The population densities for four regional disposal sites averaged over a 50-mile radius around the sites are as follows: northeast: 228 persons/mile^2, southeast: 61 persons/mile^2, western: 6 persons/mile^2, and midwest: 79 persons/mile^2. Assuming that these population densities represent rural environments and that waste transportation will be through rural as well as urban areas, the following population densities are assumed in this report:

TPOP			
	Reference Site Environments		
NE	SE	MW	SW
2280	610	790	60

These values may be compared to the average population densities as averaged over each of the five NRC regions within the contiguous United States. These are calculated as follows (Ref. 3):

NRC Region		Ave. Pop Density (persons/mile^2)
1	(NE)	300
2	(SE)	130
3	(NW)	117
4&5	(W)	37

As shown, these average population densities are all less than the population densities assumed in this report for the four reference sites.

Another comparison may be made with the assumptions for population densities made in the Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes (Ref. 34). In this statement, a demographic model is used which considers population densities for "urbanized areas" (a central city with 50,000 or more inhabitants and surrounding closely settled territory), "other urban areas" (urban populations outside the urbanized areas around the larger cities), and rural areas. Using this data base, a demographic model is formulated which assumes that individuals are located in either urban, suburban, or rural environments. An extreme density urban environment (calculations dominated by New York City) was also considered. The population densities for this data base and demographic model are given in Table C-30. Reasonable correlation appears to be made between the assumptions in this statement of the assumptions in this report, especially considering that it is very unlikely that a waste treatment or disposal facility will be located within the heart of a builtup (central city) urban area.

The parameters TDP and TDO are in units of (miles/ft)^2 and are solutions to the transportation factor TDZ as given by equations (3-8) and (3-11) in Section 3.2.2. TDP pertains to impacts to populations along the transportation route, and is solved assuming an exclusion distance (x_{min}) equal to 30 ft for the northeast, southeast, and midwest sites, and 100 ft for the southwest site. TDO pertains to impacts to populations during stopovers, and is solved assuming x_{min} equal to 10 ft. VEL is the average speed (miles/hr) of the waste transport vehicles.

Three additional parameters include WVEL, AXOQ, and FXOQ. WVEL is merely the site wind speed, in m/sec, listed in Table C-24. AXOQ is in units of yr/m^3 and denotes the atmospheric dispersion factor used for calculation of offsite impacts to an individual resulting from a dropped container accident. It is derived using equation 4-100 as discussed in Section 4.6.1. FXOQ is also in units of yr/m^3 and denotes the atmospheric dispersion factor used for calculation of offsite impacts to an individual resulting from an accidental fire in a disposal cell. It is derived using equation 4-105 in Section 4.6.2.

Table C-30. Summary of Population Density, Data Base, and Demographic Model

Population Zone	Land area (%)	Population Distribution (%)	Population Density (persons/mile^2)
A. Urbanized Area			
--Central City	0.18	31.5	10,000
--Urban fringe	0.8	26.8	1,862
--Total	0.98	58.3	3,375
B. Other Urban Areas	0.53	15.2	1,862
C. Rural Areas	98.5	26.5	15.5
D. Demographic Model Used in Statement			
--Urban	0.18	31.5	10,000
--Suburban	1.3	42.0	1,862
--Rural	98.5	26.5	15.5
--Extreme density urban*	--	--	40,000

*Calculations dominated by only one case, New York City.

The last two parameters, RI and F3, are used as discussed in Appendix D to calculate region-specific variations in pathway dose conversion factors. RI is the irrigation rate of farm crops (m^3/m^2-day) while F3 is the rate of consumption of forage by animals (kg/day).

C.7.2.2 Dependent Derived Environmental Parameters

The derived environmental parameters which are used in this report and are dependent upon the particular treatment/disposal method are summarized in Table C-31. All parameters relate to groundwater migration calculations. The terminology "symbol" refers to the variable name used for the particular parameter in the computer codes. These parameters are assumed or automatically calculated in the computer codes for the reference sites and waste disposal facilities but may also be optionally provided by the user.

Table-31. Definitions of Dependent Derived Environmental Parameters

Symbol	Scenario	Unit	Environmental Property
PRC	Groundwater migration	m/yr	Percolation coefficient
TSC1	Groundwater migration	*	Contact time fraction, percolation speed
TSC2	Groundwater migration	*	Contact time fraction, moisture content
DTTM	Groundwater migration	yrs	Saturated zone travel time between sectors
TTM	Groundwater migration	yrs	Travel time in saturated zone from first sector to appropriate biota access location
DTPC	Groundwater migration	*	Peclet number between sectors
TPC	Groundwater migration	*	Peclet number from first sector to appropriate biota access location

*Dimensionless

Specific assumed values for the first three parameters (PRC, TSC1, and TSC2) are read into the codes from a data file, and these values are listed in Table C-32. The remaining parameters are calculated within the computer codes.

The parameter PRC is the annual percolation (p) into the disposal cells, in m/yr. Two alternative methods are available to determine values for the contact time fraction (t_c or TSC). The first method (TSC1) is based on the speed of the percolating water while the second method (TSC2) is based on the average moisture content of the disposal cells. The parameter TSC1 is dimensionless and is given as the following:

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$$TSC1 = t_c = PRC/nv \tag{C-46}$$

where

t_c = PRC/nv, the contact time fraction,
PRC = percolation into waste (m/yr),
n = effective porosity of the disposed waste, and
v = speed of percolating water (m/yr).

For this work, n is assumed to be 0.25 and v is assumed to be 111 m/yr (1 ft/day). The parameter TSC2 is also dimensionless and is estimated based on some semi-quantitative observations.

The percolation, PRC, is a function of the amount of rainfall at the site (plus other site environmental conditions) as well as the assumed effectiveness of the disposal facility covers. PRC and TSC are discussed in Sections 4.3.1.1 and 4.3.1.2.

Formulation of the travel times and Peclet numbers is discussed in Section 4.3.2.4. Briefly, groundwater migration is modeled by assuming that the disposal area within the facility is laid out as a rectangle with the longer dimension conservatively aligning with the direction of groundwater flow. The disposal area (A) is calculated using equation (C-2). The dimensions of the rectangular disposal area (Figure C.17) are assumed to be given by the dimensional relationship of the golden rectangle of Greek antiquity:

$$\frac{a}{b} = \frac{b}{a-b} \tag{C-47}$$

Substituting x = a/b, it can be shown using the quadratic equation that:

$$x = \frac{a}{b} = \frac{1 + \sqrt{5}}{2} \approx 1.618 \tag{C-48}$$

Substituting this into equation (C-47), plus the identity (ab = A), it can be shown that the longitudinal dimension (a) is given by:

$$a = (A(1 + x)/x)^{\frac{1}{2}} \tag{C-49}$$

The disposal area is then divided into 10 sectors, and 1/10 of the disposal site radionuclide inventory is assumed to be released as a point source from the center of each sector. The width of each sector is thus calculated as a/10. The dose rate at each of the four downgradient biota access locations (Figure C.17) is obtained by summing the dose rate calculated from the releases from each sector. The intruder well is located at the downgradient edge of the disposal area, while the boundary well is a distance (c) from the edge of the intruder well. The code user has the option, using the IBUF index, of choosing either a 100 ft buffer zone (IBUF = 1) or a 1,000 ft buffer zone (IBUF = 2). The distances to the surface water location and to the population well are independent of the disposal area. The distance from the center of the disposal area to the surface water has been discussed in Section C.7.2.1 and is given as d = DSUR. The distance to the population well is given as d/2 = DSUR/2.

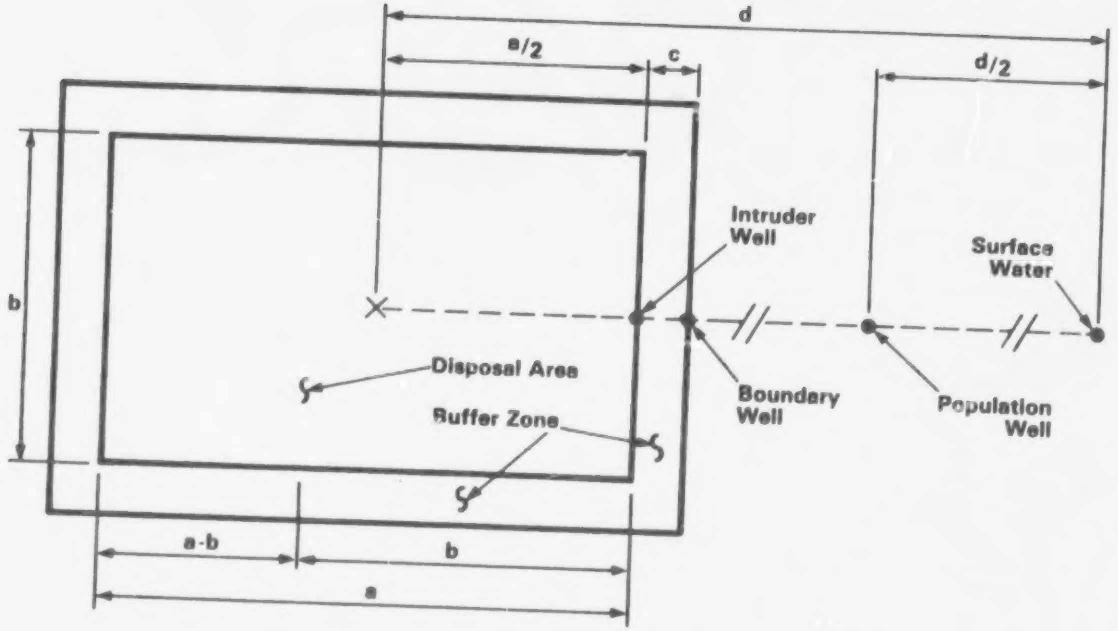
d
a/2
c
d/2
Intruder Well
Surface Water
b
Disposal Area
Buffer Zone
Boundary Well
Population Well
a-b
b
a

Figure C.17. Disposal Area Dimensions

Table C-32. Dependent Derived Environmental Parameters

Symbol	Scenario/Env. Prop.	Reference Site Environments			
		NE	SE	MW	SW
PRC(1)	Base percolation	.074	.180	.050	.001
PRC(2)	Improved percolation	.036	.030	.025	.001
TSC1	Base contact time fraction	2.66E-3	6.47E-3	1.80E-3	3.60E-5
TSC1	Improved contact time fraction	1.29E-3	1.08E-3	9.00E-4	3.60E-5
TSC2(1)	Base contact time fraction	0.3	0.3	0.3	0.2
TSC2(2)	Improved contact time fraction	0.2	0.2	0.2	0.1

The above dimensions and distances are used to determine radionuclide travel times. TTM is the time (in years) for radionuclides to travel horizontally through the saturated zone from the first sector (the sector nearest the biota access locations) to the particular biota access location considered. For an example waste disposal facility located in the southeast site, and assuming random disposal of 1,000,000 m^3 of waste by the reference shallow land burial trenches the disposal area is equal to 1,000,000/(0.5 x 6.23 x 0.88) = 366,000 m^2. The disposal area dimensions are given as a = 772 m and b = 477 m, while the sector width is equal to 77.2 m.

TTM for this example is 30.9 years to the intruder well. This corresponds to a travel time of 30.9 years across half the first sector distance (38.6 m divided by a ground water speed of 1.25 m/yr). The travel time for each of the next nine sectors is determined by adding successive multiples of the travel time between the midpoints of adjacent sectors (61.8 years for this example).

TTM is calculated for the other biota access locations in a similar manner. For example, the distance from the center of the first sector to the surface water access location is given as (d - a/2 + a/20) = (1000 - $^{772}/2$ + $^{772}/20$) = 652.6 m. TTM is thus 652.6/1.25 = 522.1 yrs.

The travel time between adjacent sectors, designated by DTTM, is determined by dividing the sector widths by the groundwater speeds.

The Peclet numbers are determined and used in an analogous manner to the groundwater travel times, where a Peclet number is calculated as the distance to the biota access location divided by the longitudinal dispersivity of the medium. The assumed dispersivities (DISP) are 0.05 m for the northeast site, 0.05 m for the southeast site, 0.05 m for the midwest site, and 0.05 m for the southwest site. TPC is the Peclet number calculated for the distance between the first sector (the sector closest to the biota access locations) and the particular biota access location. The Peclet number for each of the next nine sectors is determined by adding successive multiples of the Peclet number calculated for the distances between the midpoints of adjacent sectors. The Peclet number between sectors is as denoted as DTPC.

A summary of the distances (m), Peclet numbers, and horizontal travel times for this example problem are given below:

Distances (m)	Distances (m)	Peclet numbers	Horz. travel times (yr)
Between sectors	77.2	1544	61.8
From first sector midpoint:			
· to intruder well	38.6	772	30.9
· to boundary well*	69.1	1382	55.3
· to population well	326.3	6526	261
· to surface water	652.6	13,052	522.1

*Assuming 100 ft (30.5 m) buffer zone

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APPENDIX D

PATHWAY DOSE CONVERSION FACTORS

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APPENDIX D. Pathway Dose Conversion Factors

This appendix presents the data and calculational procedures used to determine the pathway dose conversion factors (PDCFs) used in the computer codes to determine radiological impacts resulting from the presence of a given concentration of a particular radionuclide at a particular biota access location.

D.1 CONCEPT OF PDCF

There are many diverse mechanisms through which radionuclides contained in low-level radioactive waste streams may be potentially released (e.g., mobilized from the waste and become accessible to a transport agent such as wind or water), transported through the environment (e.g., moved from one location to another through the atmosphere or soil by a transport agent), and thereby become accessible to humans through various pathways. Human access to the radioactive material may result either through direct human contact with contaminated material (e.g., inhalation of air, ingestion of water, or direct exposure to radiation) or indirectly through contaminated biota (through a multitude of pathways involving ingestion of plants or animals which have taken up radionuclides) which have come into contact with contaminated material. Each of these radionuclide release/transport/pathway combinations (scenarios) represents a complex series of interactions which are affected by a wide range of parameters such as waste properties, treatment or disposal facility environmental conditions, and operational procedures.

In this report, the generalized approach to calculating human exposures to an individual from a given release/transfer/pathway scenario and a given radioactive waste stream is given as follows:

$$H = \sum_{i=1}^{N} C_i \times I_i \times PDCF_i \qquad \text{(D-1)}$$

where:

H = the dose rate to an individual in units of mrem/yr;

C_i = the concentration of the ith radionuclide (in Ci/m^3) in the particular waste stream considered;

I_i = a transfer factor relating the concentration of the radionuclide in the waste to the concentration of the waste at a biota access location (a location, such as a well drilled into an aquifer, where humans may be exposed to or uptake radioactivity); and

$PDCF_i$ = the pathway dose conversion factor for the ith radionuclide, generally in units of mrem/yr per Ci/m^3.

In the above equation, the generalized transfer factor, I, represents the movement of a radionuclide from the waste to a biota access location. This frequently involves a complex series of calculations which are discussed in appropriate chapters of the main report. Once the concentration of a given radionuclide is determined at a given biota access location, the interaction of the

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radionuclide (and the radiation it emits) with human tissue must be quantified and the resultant dose rates determined. This is done through a series of radionuclide-specific parameters known as pathway dose conversion factors which are independent of the original means of contamination.

As an example, consider a hypothetical event in which a construction worker operates heavy machinery within an excavation into land formerly used for a low-level waste disposal facility. This hypothetical "intruder-construction scenario" is postulated to occur long after the close of the disposal facility during a temporary breakdown in institutional controls. Over the intervening years, the waste has decomposed and the excavation process stirs up a cloud of dust which contains a portion of the disposed radioactivity. Thus, the worker can be modeled as being immersed in a cloud of dust containing known concentrations (Ci per m^3 of air) of a particular group of radionuclides. Potential exposures to the worker could result from the following five pathways:

- inhalation of the contaminated air;
- direct ionizing radiation exposure from being immersed in the contaminated air;
- direct ionizing radiation exposure from contaminated dust which has settled to the ground surface;
- inhalation of contaminated dust which has been resuspended from the ground surface; and
- immersion in the contaminated dust which has been resuspended from the ground surface.

The resulting radiological impacts would be determined by a series of equations describing the movement and uptake of the radioactivity. Factors such as the breathing rate of the individual would be considered, in addition to the deposition and resuspension rates of the individual radionuclides, and so forth. A series of dose conversion factors (termed fundamental dose conversion factors (DCF) in this report to distinguish them from PDCF's) would be used to determine resulting exposures. (For example, once a known quantity of a given radionuclide is inhaled, a dose rate can be determined by multiplying by an appropriate inhalation DCF.) The total dose rate from immersion in the contaminated cloud would be determined by adding the dose rate from all five pathways.

Consider, however, that one may be interested in repeatedly calculating dose rates to different individuals at different times. In addition, the above five pathways would be appropriate for a number of situations in which an individual could be immersed in a contaminated cloud. It would be highly inconvenient to repeatedly calculate radionuclide movement and uptake through the five example pathways, and repeatedly sum the result. Furthermore, the main use of the calculational methodology is expected to be in generic rather than site-specific calculations.

Based on the above, the approach taken is to decouple the dose rate calculational procedure from the original means of contamination. Using the above example, a single PDCF is created for each radionuclide which is a combination of the impacts received through the five separate pathways. That is, given a unit concentration (1 Ci/m^3) of a radionuclide in air, the dose rates from all five pathways are summed to effectively form a single pathway dose conversion factor. Different PDCFs are created for different combinations of environmental pathways.

The pathway dose conversion factors used in this report are summarized in Tables D-1 and D-2. All PDCFs, except PDCF-8, are given in units of mrem/yr per Ci/m^3 in the media at the biota access location. Depending upon the situation, the media may consist of a solid (e.g., Ci per m^3 of soil), a liquid (e.g., Ci per m^3 of water), or a gas (e.g., Ci per m^3 of air). PDCF-8 is given in units of mrem/yr per Ci/m^3.

Different PDCFs are used appropriate to a given situation. As shown, some PDCFs are composed of primary and secondary pathways. For some exposure scenarios, more than one PDCF is used.

The first three pathway dose conversion factors are very similar and are all used to estimate exposures when the biota access location is contaminated air. In all three cases, the primary pathways involve air contaminated by the primary release mechanisms, while secondary pathways involve air contaminated as a result of resuspension of settled dust particles. PDCF-1 is used to determine impacts from airborne releases of radionuclides which occur on a chronic basis and in a working environment. Applications of PDCF-1 include calculation of impacts to personnel resulting from continuous airborne releases during operation of a waste incinerator. PDCF-2 is used to determine impacts from airborne release of radionuclides due to construction activities at a closed disposal site (intruder-construction scenario), or for other impact scenarios when the period of exposure lasts for considerably less than a year (acute exposures). The acute nature of the impacts are considered when formulating the equations used to describe the pathways.

PDCF-3 is also used to determine impacts of airborne releases due to the actions of an inadvertent intruder, but for a scenario in which the intruder is assumed to live at the closed disposal site (intruder-agriculture scenario). The period of exposure is assumed to last continuously for several years. The main difference between PDCF-1 and PDCF-3 is that the latter includes a pathway involving consumption of leafy vegetables dusted by suspended radionuclides.

PDCF-4 is used to determine chronic exposures from consumption of food (vegetables, etc.) grown in contaminated soil, as well as consumption of animals (or animal products) which have been fed forage grown in contaminated soil. It is used to calculate a portion of the impacts arising from the intruder-agriculture scenario.

PDCF-5 determines direct gamma radiation exposures due to a person's proximity to soil contaminated with radionuclides. Exposures are modeled as an infinite slab source. PDCF-5 is used extensively in the computer codes to determine direct gamma radiation exposures from applications including waste transportation and post-disposal scenarios involving inadvertent human intrusion. In the equations describing exposure to humans resulting from these applications, correction factors are inserted which correct for specific materials (which may be different from soil) and exposure geometries.

PDCF-6 and PDCF-7 are used to determine impacts due to use of contaminated water. The primary pathways include direct consumption of water (and fish in the case of PDCF-7), while secondary pathways involve use of the water as irrigation. Consumption of watered crops is considered as well as resuspension of contaminated dust particles from irrigated ground surfaces. PDCF-7 differs from PDCF-6

Table D-1. Uptake Pathways Considered for Pathway Dose Conversion Factors

PDCF	Biota Access media	Uptake Pathways	
1*	air	Inhalation (air)	(p)**
		Direct Radiation (air)	(p)
		Inhalation (soil)	(s)**
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
2*	air	Inhalation (air)	(p)
		Direct Radiation (air)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
3	air	Inhalation (air)	(p)
		Direct Radiation (air)	(p)
		Food (air)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
4	soil	Food (soil)	(p)
5	soil	Direct Radiation (Volume)	(p)
6	well water	Food (water)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
7	open water	Food (water)	(p)
		Fish (water)	(p)
		Inhalation (soil)	(s)
		Direct Radiation (area)	(s)
		Direct Radiation (air)	(s)
8	soil	Direct Radiation (area)	(p)

*PDCF-1 is used for exposures that last approximately for an entire year for several years (chronic exposures), while PDCF-2 is used for exposures that occur once for considerably less than a year (acute exposures).

**(p) = primary pathway, (s) = secondary pathway.

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Table D-2. Biota Access Location-to-Human Pathway Descriptions

Pathway Designation	Description
Food (soil)	This uptake pathway includes a total of three subpathways and denotes uptake of radionuclides originating in plants via soil-to-root transfer from contaminated soil: plant-to-human plant-to-animal-to-human plant-to-animal-to-product-to-human
Food (air)	This uptake pathway includes a total of six subpathways and includes the above three food (soil) subpathways resulting from uptake of radionuclides originating on plant surfaces via deposition from contaminated air and the same three food (soil) subpathways resulting from fallout contamination of the ground.
Food (water)	This uptake pathway includes a total of nine subpathways and includes all the food (soil) pathways resulting from radionuclides originating on plant surfaces via irrigation deposition from contaminated water and from irrigation contamination of the ground. The following three subpathways in addition to the plant pathways are added: water-to-human water-to-animal-to-human water-to-animal-to-product-to-human
Ingestion (fish)	Uptake of radionuclides from eating fish caught in contaminated open water.
Inhalation (air)	Uptake of radionuclides from breathing air contaminated due to suspension of contaminated soil particulates by human activities.
Inhalation (soil)	Uptake of radionuclides from breathing air contaminated due to natural suspension and volatilization of surface soil.
Direct Radiation (volume)	Direct exposure to ionizing radiation from standing on ground homogeneously contaminated.
Direct Radiation (area)	Direct exposure to ionizing radiation from standing on ground whose surface is contaminated.
Direct Radiation (air)	Direct exposure to ionizing radiation from standing in air homogeneously contaminated.

in that PDCF-7 is used for exposures involving open water bodies such as a stream while PDCF-6 is used for well water applications.

PDCF-8 is the only PDCF having units of mrem/yr per Ci/m^2, and has been included to calculate impacts resulting from a specific scenario involving inadvertent intrusion into a closed disposal facility. This involves radiological impacts due to proximity to a sealed source potentially imbedded in the wall of an excavation.

The PDCFs used in this report are generic. That is, they are constructed assuming a set of parameters typical of many environmental conditions and human actions. Different assumptions could be incorporated depending upon site-specific environmental conditions. In addition, all PDCFs are calculated assuming that the exposure period lasts for a total year. Many exposures, however, are acute and last for considerably less than a year. For such acute exposures a correction factor used to normalize acute exposures to the one-year reference value is incorporated into the calculations. This correction factor is generally termed the exposure duration factor (EDF) in the main body of this report.

D.2 CALCULATIONAL PROCEDURES

PDCFs are calculated based upon a number of sources. Equations used to calculate the PDCFs are mainly based upon references 1 and 2, and generally adopt the following generic form:

$$PDCF_{irps} = \sum_{p=1}^{N} f_{ips}\, DCF_{irp} \qquad \text{(D-2)}$$

where:

$PDCF_{irps}$ = the pathway dose conversion factor (50-year dose commitment in mrem/year per unit radionuclide concentration (Ci/m^3) at the biota access location) specific to organ (r), radionuclide (i), pathway (p) and scenario (s);

N = the total number of pathways in the scenario;

f_{ips} = the pathway usage factor of nuclide (i) of pathway (p) via scenario (s) which is considered in the calculation of the accumulated radiation dose to man (generally in units of m^3/yr); and

DCF_{irp} = the fundamental dose conversion factor, a value specific to a given radionuclide (i), pathway (p), and organ (r) which is used to calculate radiation dose commitments (generally in units of mrem/pCi).

The specific equations used to calculate the PDCFs are summarized in Table D-3. In the table, the components of each equation are first separated into each individual pathway comprising the PDCF, and then the equation for the PDCF is summarized.

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Table D-3. Equations Used to Calculate Pathway Dose Conversion Factors

PDCF	Uptake Pathways	Pathway Equations
1	Inhalation (air)	$C*f_{15}*DCF2$
	Direct Radiation (air)	$C*DCF5$
	Inhalation (soil)	$C*D_1*f_{18}*f_{14}*f_{15}*DCF2$
	Direct Radiation (area)	$C*D_1*f_{18}*DCF4$
	Direct Radiation (air)	$C*D_1*f_{18}*f_{14}*DCF5$

$$\begin{aligned} PDCF\text{-}1 = {} & C*(f_{15}*DCF2 + DCF5) \\ & + C*D_1*f_{18}*(f_{14}*(f_{15}*DCF2 + DCF5) + DCF4) \end{aligned} \qquad \text{(D-3)}$$

PDCF	Uptake Pathways	Pathway Equations
2	Inhalation (air)	$C*f_{15}*DCF2$
	Direct Radiation (air)	$C*DCF5$
	Inhalation (soil)	$C*D_1*f_{18}*f_{14}*f_{15}*DCF2*0.242$
	Direct Radiation (area)	$C*D_1*f_{18}*DCF4*0.242$
	Direct Radiation (air)	$C*D_1*f_{18}*f_{14}*DCF5*0.242$

$$\begin{aligned} PDCF\text{-}2 = {} & C*(f_{15}*DCF2 + DCF5) \\ & + C*D_1*f_{18}*(f_{14}*(f_{15}*DCF2 + DCF5) + DCF4)*0.242 \end{aligned} \qquad \text{(D-4)}$$

PDCF	Uptake Pathways	Pathway Equations
3	Inhalation (air)	$C*f_{15}*DCF2$
	Direct Radiation (air)	$C*DCF5$
	Food (air)	$C*(D_1*PT + (D_2/CY)*PTP)*DCF1$
	Inhalation (soil)	$C*D_1*f_{18}*f_{14}*f_{15}*DCF2$
	Direct Radiation (area)	$C*D_1*f_{18}*DCF4$
	Direct Radiation (air)	$C*D_1*f_{18}*f_{14}*DCF5$

$$\begin{aligned} PDCF\text{-}3 = {} & C*(f_{15}*DCF2 + DCF5) \\ & + C*(D_1*PT + D_2/CY)*F*PTP)*DCF1 \\ & + C*D_1*f_{18}*(f_{14}*(f_{15}*DCF2 + DCF5) + DCF4) \end{aligned} \qquad \text{(D-5)}$$

Table D-3 (Continued)

PDCF	Uptake Pathways	Pathway Equations
4	Food (soil)	$C*(PT/D)*DCF1$

$$PDCF\text{-}4 = C*(PT/D)*DCF1 \tag{D-6}$$

5	Direct Radiation (Volume)	$C*DCF3$

$$PDCF\text{-}5 = C*DCF3 \tag{D-7}$$

6	Food (water)	$C*(W_1*PT + (W_2/CY)*PTP + FT/1000)*DCF1$
	Inhalation (soil)	$C*W_1*f_{18}*f_{14}*f_{15}*DCF2$
	Direct Radiation (area)	$C*W_1*f_{18}*DCF4$
	Direct Radiation (air)	$C*W_1*f_{18}*f_{14}*DCF5$

$$\begin{aligned} PDCF\text{-}6 = {} & C(W_1*PT + (W_2/CY)*PTP + FT/1000)*DCF1 \\ & + C*W_1*F_{18}* (f_{14}*(f_{15}*DCF2 + DCF5) + DCF4) \end{aligned} \tag{D-8}$$

7	Food (water)	$C*(W_1PT + (W_2/CY)*PTP + FT/1000)*DCF1$
	Fish (water)	$C*(F_{12}/1000)*DCF1$
	Inhalation (soil)	$C*W_1*f_{18}*f_{14}*f_{15}*DCF2$
	Direct Radiation (area)	$C*W_1*f_{18}*DCF4$
	Direct Radiation (air)	$C*W_1*f_{18}*f_{14}*DCF5$

$$\begin{aligned} PDCF\text{-}7 = {} & C(W_1*PT + (W_2/CY)*PTP + FT/1000)*DCF1 \\ & + C(F_{12}/1000)*DCF1 \\ & + C*W_1*f_{18}* \ (f_{14}* (f_{15}*DCF2 + DCF5) + DCF4 \end{aligned} \tag{D-9}$$

8	Direct Radiation (area)	$C*DCF4$

$$PDCF\text{-}8 = C*DCF4$$

PDCFs give a dose rate in mrem/yr per unit concentration of a radionuclide in the media of a given biota access location. The exposure media considered for each PDCF is summarized below:

PDCF	Unit Radionuclide concentration
1	1 Ci per m^3 of air
2	1 Ci per m^3 of air
3	1 Ci per m^3 of air
4	1 Ci per m^3 of soil
5	1 Ci per m^3 of soil
6	1 Ci per m^3 of water
7	1 Ci per m^3 of water
8	1 Ci per m^2 of surface

The parameters making up the equations, including fundamental dose conversion factors and translocation parameters, are discussed below following a summary of the radionuclides considered in this report.

D.2.1 Radionuclides Considered

A total of 53 radionuclides are considered in this report, and these radionuclides are listed in Table D-4, along with the radionuclide half-lives and solubility classes.

Table D-5 lists all the elemental solubility classes considered in this report (100 total), as well as typical chemical forms appropriate to each solubility class. The information on chemical forms was obtained from reference 3.

D.2.2 Fundamental Dose Conversion Factors

All the PDCFs are based upon five sets of adult fundamental dose conversion factors (DCF). These include DCFs for 50-year committed inhalation dose in units of mrem per pCi inhaled, 50-year committed ingestion dose in units of mrem per pCi ingested, and three sets of DCFs for direct radiation exposure applications. The use of these last three DCFs depends upon the particular biota access location considered, and include factors for volume contamination of soil (mrem/yr per pCi/m^3), surface contamination of soil or other material (mrem/yr per pCi/m^2), and air contamination (mrem/yr per pCi/m^3). The values of these fundamental DCFs are a function of the radionuclide of concern, the solubility class, and the organ receiving the dose. The DCFs are given in terms of nine organs (whole body, red bone marrow, bone surface (endosteal cells), liver, thyroid, kidney, lung, stomach wall, lower large intestine) plus the effective whole body equivalent as calculated using ICRP-26 and ICRP-30 methodology. These DCFs are listed in Table D-6 through D-10 according to the following schedule:

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Table D-4. Radionuclides Considered in Report

Nuclide	Solubilities	Half-Life (Years)	Radiation Emitted
H-3	*[1]	1.23E+01[2]	b[3]
C-14	*	5.73E+03	b
Na-22	D	2.62E+00	b,g
Cl-36	D,W	3.08E+05	b,g
Fe-55	W,Y	2.60E+00	b,g
Co-60	W,Y	5.26E+00	b
Ni-59	D,W	8.00E+04	b,g
Ni-63	D,W	9.20E+01	b,g
Sr-90	D,Y	2.81E+01	b,g
Nb-94	W,Y	2.00E+04	a,b,g
Tc-99	D,W	2.12E+05	b
Ru-106	Y	1.01E+00	b
Ag-108m	D,W,Y	1.27E+02	g
Cd-109	D,W,Y	1.24E+00	b,g
Sn-126	D,W	1.05E+05	g
Sb-125	D,W	2.71E+00	b,g
I-129	D	1.17E+07	b,g
Cs-134	D	2.05E+00	b,g
Cs-135	D	3.00E+06	b
Cs-137	D	3.00E+01	b,g
Eu-152	W	1.27E+01	b,g
Eu-154	W	1.60E+01	b,g
Pb-210	W	2.04E+01	a,b,g
Rn-222	*	1.05E-02	a,g
Ra-226	W	1.60E+03	a,b,g
Ra-228	W	6.70E+00	b
Ac-227	W,Y	2.16E+01	a,g
Th-228	W,Y	1.91E+00	a,g
Th-229	W,Y	7.34E+03	a,g
Th-230	W,Y	8.00E+04	a,g
Th-232	W,Y	1.41E+10	a,g
Pa-231	W,Y	3.25E+04	a,b,g
U-232	D,W,Y	7.20E+01	a,g
U-233	D,W,Y	1.62E+05	a,g
U-234	D,W,Y	2.47E+05	a,g
U-235	D,W,Y	7.10E+08	a,g
U-236	D,W,Y	2.39E+07	a,g
U-238	D,W,Y	4.51E+09	a,g
Np-237	W,Y	2.14E+06	a,g
Pu-236	W,Y	2.85E+00	a,g
Pu-238	W,Y	8.64E+01	a,g
Pu-239	W,Y	2.44E+04	a,g
Pu-240	W,Y	6.58E+03	a,g
Pu-241	W,Y	1.32E+01	a,b,g
Pu-242	W,Y	3.79E+05	a
Pu-244	W,Y	7.60E+07	a
Am-241	W,Y	4.58E+02	a,g
Am-243	W,Y	7.95E+03	a,g
Cm-242	W,Y	4.45E-01	a,g
Cm-243	W,Y	3.20E+01	a,g
Cm-244	W,Y	1.76E+01	a,g
Cm-248	W,Y	4.70E+05	a,g
Cf-252	W,Y	2.65E+00	a,g

Notes: (1) Solubility class: * - Not applicable, D - Day, W - Week, Y - Year
(2) Exponential Notation: 1.23E+01 = 1.23×10^1
(3) Radiation emitted: a - Alpha, b - Beta, g - Gamma

Table D-5. Element Solubility Classes

Element	Solubility Class	Compounds
H	*	All compounds
C	*	All compounds
Na	D	All compounds
Cl	W	Information not provided
	D	Information not provided
Fe	W	Oxides, hydroxides and halides
	D	All other compounds
Co	Y	Oxides, hydroxides, halides and nitrates
	W	All other compounds
Ni	W	Oxides, hydroxides, halides, nitrates and carbides
	D	All other compounds
Sr	Y	Strontium titanate
	W	All other compounds
Nb	Y	Oxides and hydroxides
	W	All other compounds
Tc	W	Oxides, hydroxides, halides and nitrates
	D	All other compounds
Ru	Y	Oxides and hydroxides
	W	Halides
	D	All other compounds
Ag	Y	Oxides and hydroxides
	W	Nitrates and sulphides
	D	Silver metal and all other compounds
Cd	Y	Oxides and hydroxides
	W	Sulphides, halides and nitrates
	D	All other compounds
Sn	W	Sulphides, oxides, hydroxides, halides, nitrates and stannic compounds
	D	All other compounds
Sb	W	Oxides, hydroxides, sulphides, sulphates, carbonates, nitrates, and halides
	D	All other compounds
I	D	All compounds
Cs	D	All compounds
Eu	Y	Oxides, hydroxides, carbides and fluorides
	W	All other compounds
Pb	D	All compounds
Po	W	Oxides, hydroxides and nitrates
	D	All other compounds
Rn	*	All compounds
Ra	W	All compounds
Ac	W	All compounds
Th	Y	Oxides and hydroxides
	W	All other compounds

Table D-5 (Continued)

Element	Solubility Class	Compounds
Pa	W	All compounds
U	D	Most hexavalent uranium compounds
	W	Most tetravalent uranium compounds
	Y	UO_2 and U_3O_8
Np	W	All compounds
Pu	Y	Plutonium dioxide
	W	All other compounds
Am	W	All compounds
Cm	W	All compounds
Cf	Y	Oxides and hydroxides
	W	All other compounds

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Symbol	Definition	Units	Table
DCF1	Fundamental DCF for Ingestion	mrem/yr per pCi	D-6
DCF2	Fundamental DCF for Inhalation	mrem/yr per pCi	D-7
DCF3	Fundamental DCF for External Exposure (Volume Source)	mrem/yr per pCi/m^3	D-8
DCF4	Fundamental DCF for External Exposure (Area Source)	mrem/yr per pCi/m^2	D-9
DCF5	Fundamental DCF for External Exposure (Air Immersion)	mrem/yr per pCi/m^3	D-10

A brief description of each fundamental DCF follows. All DCFs are applicable to adults.

Ingestion DCF (DCF1). Ingestion dose conversion factors (50-year committed dose) for this report were principally obtained from reference 4. DCFs for some radionuclides of interest, however, were not listed in reference 4, and so DCFs for these nuclides were filled in based upon information obtained from references 5 and 6. All ingestion DCFs are based upon more recent models for radionuclide transfer within the human body as presented by the International Commission on Radiological Protection (ICRP) in ICRP-30 (Ref. 7). A quality factor of 20 is assumed for alpha-emitting radionuclides.

Inhalation DCF (DCF2). The inhalation fundamental dose conversion factors (50-year committed dose) are based on a model originally published by the ICRP Task Group on Lung Dynamics in 1966 (Ref. 8). More recent versions of the basic model, however, are incorporated based upon ICRP-30. Similarly to the ingestion of DCFs, reference 4 was used as a source for most of the 85 radionuclides considered. DCFs for radionuclides not listed in reference 4 were obtained by consulting references 5 and 6. A median aerodynamic diameter of one micron is assumed for particulates, and a quality factor of 20 is assumed for alpha-emitting radionuclides.

Direct radiation (volume) DCF (DCF3). DCF3 is used to estimate exposures due to gamma radiation emitted by radionuclides homogeneously distributed through soil, and in which the contaminated soil extends horizontally and downward to infinity (i.e., an infinite slab source). This dose conversion factor has been developed from exposure rate data presented in HASL-195 for K-40, natural uranium, and thorium plus daughters uniformly distributed in soil (Ref. 9). Table 2 of reference 9 presents the exposure rate data as a function of gamma energy and height above soil, and this table has been used to construct a graph of exposure rate as a function of gamma energy. This graph is presented as Figure D.1 for a distance of one meter above the soil surface. Use of the graph allows the effects of self-shielding and buildup to be intrinsically considered in the calculations.

TABLE D-6 . DCF1: Fundamental Dose Conversion Factor For Ingestion

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	8.36-08[#]	1.08-07	1.43-07	8.30-08	8.56-08	8.28-08	8.26-08	6.56-08	8.28-08	8.98-08
C-14	*	8.49-07	1.21-06	1.46-06	1.92-06	1.06-06	1.23-06	3.38-06	7.05-06	8.89-07	1.54-06
NA-22	D	2.57-05	1.04-05	1.35-05	1.36-05	1.49-05	1.40-05	1.95-05	2.50-05	1.21-05	1.86-05
CL-36	D	2.96-06	4.11-06	2.96-06	3.06-06	2.96-06	2.96-06	2.96-06	2.96-06	2.96-06	3.03-06
CL-36	W	2.96-06	4.11-06	2.96-06	3.06-06	2.96-06	2.96-06	2.96-06	2.96-06	2.96-06	3.03-06
FE-55	W	7.10-07	1.84-08	7.08-07	3.50-07	6.44-07	1.22-06	6.41-07	4.37-07	6.32-07	7.79-07
FE-55	Y	7.10-07	1.84-08	7.08-07	3.50-07	6.44-07	1.22-06	6.41-07	4.37-07	6.32-07	7.79-07
CO-60	W	8.62-06	5.30-06	4.02-05	4.37-06	5.67-06	6.83-06	5.42-06	3.99-06	3.10-06	1.13-05
CO-60	Y	8.62-06	5.30-06	4.02-05	4.37-06	5.67-06	6.83-06	5.42-06	3.99-06	3.10-06	1.13-05
NI-59	D	1.30-07	1.52-07	9.99-07	2.10-07	1.32-07	1.32-07	1.35-07	1.34-07	1.44-07	2.10-07
NI-59	W	1.30-07	1.52-07	9.99-07	2.10-07	1.32-07	1.32-07	1.35-07	1.34-07	1.44-07	2.10-07
NI-63	D	3.15-07	3.89-07	3.40-06	5.77-07	3.15-07	3.15-07	3.15-07	3.15-07	3.15-07	5.77-07
NI-63	W	3.15-07	3.89-07	3.40-06	5.77-07	3.15-07	3.15-07	3.15-07	3.15-07	3.15-07	5.77-07
SR-90	D	5.99-06	8.76-07	7.78-05	9.45-05	5.99-06	5.71-06	4.30-04	8.60-04	5.99-06	8.75-05
SR-90	Y	3.00-07	8.76-07	9.60-05	5.02-06	3.00-07	2.86-07	2.15-05	4.30-05	3.00-07	1.13-05
NB-94	W	6.36-07	2.85-06	4.63-05	7.14-06	2.57-06	1.04-06	2.73-06	2.83-06	4.55-07	7.14-06
NB-94	Y	6.36-07	2.85-06	4.63-05	7.14-06	2.57-06	1.04-06	2.73-06	2.83-06	4.55-07	7.14-06
TC-99	D	3.17-07	9.30-07	3.20-06	2.14-07	4.58-07	6.28-07	3.22-07	4.10-07	1.41-05	1.02-06
TC-99	W	3.17-07	9.30-07	3.20-06	2.14-07	4.58-07	6.28-07	3.22-07	4.10-07	1.41-05	1.02-06
RU-106	Y	8.51-06	6.41-06	2.60-04	5.94-06	8.25-06	8.27-06	8.31-06	9.57-06	8.06-06	2.88-05
AG-108	D	2.23-06	3.85-06	2.84-05	7.62-06	3.92-06	2.55-05	2.45-06	1.31-06	4.81-07	7.62-06
AG-108	W	2.23-06	3.85-06	2.84-05	7.62-06	3.92-06	2.55-05	2.45-06	1.31-06	4.81-07	7.62-06
AG-108	Y	2.23-06	3.85-06	2.84-05	7.62-06	3.92-06	2.55-05	2.45-06	1.31-06	4.81-07	7.62-06
CD-109	D	1.17-06	1.51-06	1.71-05	1.82-05	1.51-04	2.73-05	1.37-06	1.21-06	1.02-06	1.31-05
CD-109	W	1.17-06	1.51-06	1.71-05	1.82-05	1.51-04	2.73-05	1.37-06	1.21-06	1.02-06	1.31-05
CD-109	Y	1.17-06	1.51-06	1.71-05	1.82-05	1.51-04	2.73-05	1.37-06	1.21-06	1.02-06	1.31-05
SN-126	D	2.22-06	5.92-06	1.60-04	1.95-05	3.03-06	2.53-06	1.01-05	1.87-05	2.04-06	1.95-05
SN-126	W	2.22-06	5.92-06	1.60-04	1.95-05	3.03-06	2.53-06	1.01-05	1.87-05	2.04-06	1.95-05
SB-125	D	2.23-07	1.10-06	2.14-05	2.81-06	4.18-07	9.21-07	8.36-07	2.17-06	1.71-07	2.81-06
SB-125	W	5.03-08	9.51-07	2.33-05	2.80-06	2.43-07	2.32-07	4.48-07	3.35-07	2.06-08	2.80-06
I-129	D	8.21-07	7.84-08	6.70-08	3.18-06	7.02-07	7.24-07	9.42-07	8.79-07	7.80-03	2.34-04
CS-134	D	1.60-04	4.99-05	5.75-05	6.84-05	1.00-04	1.00-04	9.26-05	8.86-05	7.81-05	1.12-04
CS-135	D	1.12-05	2.40-07	5.35-07	6.61-06	1.12-05	1.12-05	1.12-05	1.30-05	1.13-05	1.13-05
CS-137	D	1.00-04	2.18-05	2.59-05	4.91-05	7.73-05	7.87-05	7.38-05	7.99-05	6.72-05	8.19-05
EU-152	W	8.88-07	2.38-06	3.70-05	6.48-06	1.72-06	1.11-05	3.40-06	7.73-06	2.46-07	6.48-06
EU-154	W	7.99-07	3.03-06	6.66-05	9.55-06	1.66-06	1.37-05	4.26-06	1.65-05	2.11-07	9.55-06
PB-210	W	3.00-04	1.95-07	2.03-05	1.70-03	9.40-04	1.40-03	1.00-03	9.60-03	3.00-04	7.77-04
AC-227	Y	1.10-04	3.73-07	2.48-05	1.30-03	2.00-03	1.60-02	1.00-02	1.00-01	1.10-04	5.52-03
AC-227	W	1.10-04	3.73-07	2.48-05	1.30-03	2.00-03	1.60-02	1.00-02	1.00-01	1.10-04	5.52-03
TH-228	Y	7.50-06	4.91-06	4.70-04	3.80-05	7.88-06	2.34-05	3.80-04	4.10-03	7.42-06	2.10-04
TH-228	W	7.50-06	4.91-06	4.70-04	3.80-05	7.88-06	2.34-05	3.80-04	4.10-03	7.42-06	2.10-04
TH-229	Y	1.01-05	4.88-06	2.10-04	2.30-04	9.44-06	3.64-05	2.60-03	3.20-02	9.39-06	1.29-03
TH-229	W	1.01-05	4.88-06	2.10-04	2.30-04	9.44-06	3.64-05	2.60-03	3.20-02	9.39-06	1.29-03
RN-222	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
RA-226	W	5.90-04	5.37-06	3.30-04	3.40-03	5.90-04	5.90-04	2.20-03	2.00-02	5.90-04	1.38-03
RA-228	W	4.10-04	2.34-06	7.14-05	1.70-03	4.00-04	4.00-04	1.30-03	9.80-03	4.00-04	7.89-04
TH-230	Y	4.56-06	4.03-06	1.80-04	9.24-05	4.31-06	2.18-05	1.00-03	1.60-02	4.56-06	6.29-04
TH-230	W	4.56-06	4.03-06	1.80-04	9.24-05	4.31-06	2.18-05	1.00-03	1.60-02	4.56-06	6.29-04
TH-232	Y	3.97-06	3.47-06	1.50-04	9.63-05	3.75-06	1.88-05	1.10-03	1.80-02	3.94-06	6.92-04
TH-232	W	3.97-06	3.47-06	1.50-04	9.63-05	3.75-06	1.88-05	1.10-03	1.80-02	3.94-06	6.92-04

[#] Note: 8.36-08 means 8.36 x 10^{-8}

TABLE D-6 . DCF1: Fundamental Dose Conversion Factor For Ingestion

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	2.20-04	4.70-06	1.90-04	2.10-03	3.60-03	2.80-02	1.50-02	1.78-01	2.20-04	9.55-03
PA-231 W	2.20-04	4.70-06	1.90-04	2.10-03	3.60-03	2.80-02	1.50-02	1.78-01	2.20-04	9.55-03
U-232 Y	1.30-06	4.66-06	2.00-04	8.40-05	1.30-04	1.15-06	7.10-04	7.80-03	1.21-06	3.45-04
U-232 W	3.24-05	6.42-06	2.00-04	2.10-03	3.30-03	2.86-05	1.80-02	1.95-01	3.02-05	8.19-03
U-232 D	3.24-05	6.42-06	2.00-04	2.10-03	3.30-03	2.86-05	1.80-02	1.95-01	3.02-05	8.19-03
U-233 Y	6.95-07	4.18-06	1.80-04	2.40-05	6.78-05	6.06-07	9.60-06	1.40-04	6.39-07	2.53-05
U-233 W	1.74-05	5.51-06	1.70-04	5.80-04	1.70-03	1.51-05	2.40-04	3.50-03	1.60-05	2.61-04
U-233 D	1.74-05	5.51-06	1.70-04	5.80-04	1.70-03	1.51-05	2.40-04	3.50-03	1.60-05	2.61-04
U-234 Y	6.87-07	4.17-06	1.80-04	2.38-05	6.69-05	6.32-07	9.34-06	1.40-04	6.32-07	2.51-05
U-234 W	1.72-05	5.48-06	1.70-04	5.80-04	1.70-03	1.58-05	2.30-04	3.50-03	1.58-05	2.57-04
U-234 D	1.72-05	5.48-06	1.70-04	5.80-04	1.70-03	1.58-05	2.30-04	3.50-03	1.58-05	2.57-04
U-235 Y	7.44-07	4.05-06	1.90-04	2.16-05	6.05-05	6.08-07	7.38-06	1.20-04	5.80-07	2.44-05
U-235 W	1.62-05	5.37-06	1.80-04	5.20-04	1.50-03	1.38-05	1.80-04	2.90-03	1.45-05	2.24-04
U-235 D	1.62-05	5.37-06	1.80-04	5.20-04	1.50-03	1.38-05	1.80-04	2.90-03	1.45-05	2.24-04
U-236 Y	6.47-07	3.92-06	1.70-04	2.24-05	6.32-05	5.95-07	8.03-06	1.30-04	5.95-07	2.35-05
U-236 W	1.62-05	5.16-06	1.60-04	5.40-04	1.60-03	1.49-05	2.00-04	3.10-03	1.49-05	2.37-04
U-236 D	1.62-05	5.16-06	1.60-04	5.40-04	1.60-03	1.49-05	2.00-04	3.10-03	1.49-05	2.37-04
U-238 Y	6.14-07	3.67-06	1.70-04	2.12-05	5.96-05	5.35-07	7.65-06	1.10-04	5.63-07	2.24-05
U-238 W	1.53-05	4.85-06	1.60-04	5.10-04	1.50-03	1.34-05	1.90-04	2.80-03	1.41-05	2.20-04
U-238 D	1.53-05	4.85-06	1.60-04	5.10-04	1.50-03	1.34-05	1.90-04	2.80-03	1.41-05	2.20-04
NP-237 Y	1.20-04	5.24-06	2.00-04	1.00-03	2.10-03	1.60-02	6.10-03	8.70-02	1.20-04	4.69-03
NP-237 W	1.20-04	5.24-06	2.00-04	1.00-03	2.10-03	1.60-02	6.10-03	8.70-02	1.20-04	4.69-03
PU-236 Y	3.84-07	4.96-06	2.20-04	4.48-06	8.13-06	6.28-05	2.08-05	2.40-04	3.84-07	3.31-05
PU-236 W	3.84-07	4.96-06	2.20-04	4.48-06	8.13-06	6.28-05	2.08-05	2.40-04	3.84-07	3.31-05
PU-238 Y	3.23-06	4.71-06	2.10-04	2.83-05	5.72-05	4.40-04	1.70-04	2.10-03	3.23-06	1.37-04
PU-238 W	3.23-06	4.71-06	2.10-04	2.83-05	5.72-05	4.40-04	1.70-04	2.10-03	3.23-06	1.37-04
PU-239 Y	3.63-06	4.42-06	2.00-04	3.13-05	6.33-05	4.90-04	1.90-04	2.60-03	3.63-06	1.57-04
PU-239 W	3.63-06	4.42-06	2.00-04	3.13-05	6.33-05	4.90-04	1.90-04	2.60-03	3.63-06	1.57-04
PU-240 Y	3.62-06	4.42-06	2.00-04	3.13-05	6.32-05	4.90-04	1.90-04	2.60-03	3.62-06	1.57-04
PU-240 W	3.62-06	4.42-06	2.00-04	3.13-05	6.32-05	4.90-04	1.90-04	2.60-03	3.62-06	1.57-04
PU-241 Y	7.51-08	2.23-08	9.92-07	6.19-07	1.23-06	9.50-06	3.86-06	4.83-05	7.49-08	2.79-06
PU-241 W	7.51-08	2.23-08	9.92-07	6.19-07	1.23-06	9.50-06	3.86-06	4.83-05	7.49-08	2.79-06
PU-242 Y	3.45-06	4.21-06	1.90-04	2.98-05	6.01-05	4.70-04	1.80-04	2.60-03	3.45-06	1.53-04
PU-242 W	3.45-06	4.21-06	1.90-04	2.98-05	6.01-05	4.70-04	1.80-04	2.60-03	3.45-06	1.53-04
PU-244 Y	3.71-06	4.68-06	3.00-04	3.01-05	6.00-05	4.60-04	1.70-04	2.60-03	3.44-06	1.62-04
PU-244 W	3.71-06	4.68-06	3.00-04	3.01-05	6.00-05	4.60-04	1.70-04	2.60-03	3.44-06	1.62-04
AM-241 Y	1.20-04	4.85-06	2.10-04	1.00-03	2.20-03	1.70-02	6.40-03	8.00-02	1.20-04	4.59-03
AM-241 W	1.20-04	4.85-06	2.10-04	1.00-03	2.20-03	1.70-02	6.40-03	8.00-02	1.20-04	4.59-03
AM-243 Y	1.30-04	5.35-06	2.20-04	1.00-03	2.20-03	1.70-02	6.40-03	8.50-02	1.30-04	4.73-03
AM-243 W	1.30-04	5.35-06	2.20-04	1.00-03	2.20-03	1.70-02	6.40-03	8.50-02	1.30-04	4.73-03
CM-242 Y	2.72-06	5.25-06	2.30-04	2.58-05	5.72-05	4.40-04	1.40-04	1.50-03	2.72-06	1.18-04
CM-242 W	2.72-06	5.25-06	2.30-04	2.58-05	5.72-05	4.40-04	1.40-04	1.50-03	2.72-06	1.18-04
CM-243 Y	8.30-05	5.92-06	2.50-04	7.10-04	1.50-03	1.20-02	4.20-03	4.80-02	8.22-05	2.92-03
CM-243 W	8.30-05	5.92-06	2.50-04	7.10-04	1.50-03	1.20-02	4.20-03	4.80-02	8.22-05	2.92-03
CM-244 Y	6.40-05	4.96-06	2.20-04	5.60-04	1.20-03	9.30-03	3.30-03	3.80-02	6.40-05	2.31-03
CM-244 W	6.40-05	4.96-06	2.20-04	5.60-04	1.20-03	9.30-03	3.30-03	3.80-02	6.40-05	2.31-03
CM-248 Y	5.80-04	7.80-05	1.10-03	4.00-03	8.10-03	6.30-02	6.10-03	8.80-02	4.80-04	8.70-03
CM-248 W	5.80-04	7.80-05	1.10-03	4.00-03	8.10-03	6.30-02	6.10-03	8.80-02	4.80-04	8.70-03
CF-252 Y	3.28-05	2.10-05	5.70-04	2.30-04	5.20-04	4.00-03	6.50-04	6.90-03	2.51-05	6.66-04
CF-252 W	3.28-05	2.10-05	5.70-04	2.30-04	5.20-04	4.00-03	6.50-04	6.90-03	2.51-05	6.66-04

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TABLE D-7 . DCF2: Fundamental Dose Conversion Factor For Inhalation

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	1.25-07[#]	1.25-07	1.33-07	1.25-07	1.29-07	1.24-07	1.24-07	9.85-08	1.24-07	1.26-07
C-14	*	6.18-09	7.35-09	7.22-09	1.41-08	7.92-09	8.88-09	2.42-08	5.08-08	6.48-09	1.09-08
NA-22	D	9.43-06	6.22-06	7.53-06	9.16-06	9.91-06	9.45-06	1.30-05	1.60-05	8.14-06	1.14-05
CL-36	D	4.92-06	2.04-06	1.86-06	2.41-06	1.86-06	1.86-06	1.86-06	1.86-06	1.86-06	2.24-06
CL-36	W	1.69-04	2.46-06	1.86-06	3.12-05	1.86-06	1.86-06	1.86-06	1.86-06	1.86-06	2.19-05
FE-55	W	3.20-06	1.04-08	3.64-07	6.29-07	1.09-06	2.07-06	1.08-06	7.40-07	1.07-06	1.57-06
FE-55	Y	2.51-05	1.28-08	3.91-07	6.09-07	5.05-07	9.58-07	5.03-07	3.43-07	4.95-07	3.96-06
CO-60	W	1.30-04	1.82-05	2.78-05	1.71-05	1.87-05	2.90-05	1.75-05	1.47-05	1.44-05	3.59-05
CO-60	Y	1.30-03	1.00-04	2.85-05	8.20-05	5.85-05	1.20-04	6.45-05	5.08-05	6.01-05	2.41-04
NI-59	D	1.33-06	1.27-06	1.42-06	1.32-06	1.28-06	1.28-06	1.31-06	1.30-06	1.39-06	1.32-06
NI-59	W	4.44-06	3.92-07	8.29-07	9.19-07	3.85-07	3.85-07	3.92-07	3.89-07	4.22-07	9.18-07
NI-63	D	3.23-06	3.05-06	3.52-06	3.10-06	3.05-06	3.04-06	3.04-06	3.04-06	3.04-06	3.10-06
NI-63	W	1.14-05	9.51-07	2.49-06	2.30-06	9.18-07	9.14-07	9.14-07	9.14-07	9.14-07	2.30-06
SR-90	D	9.89-06	1.97-07	1.41-05	2.40-04	1.46-05	1.46-05	1.10-03	2.20-03	1.46-05	2.10-04
SR-90	Y	8.50-03	2.86-05	8.90-04	1.50-04	3.65-06	1.90-05	1.20-04	2.30-04	3.65-06	1.45-03
NB-94	W	1.55-04	1.31-05	3.07-05	3.61-05	2.56-05	1.39-05	2.36-05	3.37-05	9.73-06	3.61-05
NB-94	Y	2.77-03	1.18-04	3.49-05	4.14-04	7.10-05	1.45-04	8.36-05	7.29-05	8.21-05	4.14-04
TC-99	D	9.63-07	4.90-07	5.09-07	1.69-07	3.48-07	4.77-07	2.44-07	3.11-07	1.07-05	7.08-07
TC-99	W	5.22-05	5.70-07	1.66-06	8.87-07	3.07-07	4.21-07	2.15-07	2.75-07	9.46-06	7.25-06
RU-106	Y	3.80-03	6.96-06	1.40-04	6.18-05	8.95-06	1.15-05	9.37-06	1.00-05	9.19-06	5.22-04
AG-108	D	2.22-05	1.77-05	1.05-05	3.01-05	3.04-05	2.41-04	1.14-05	8.62-06	4.59-06	3.01-05
AG-108	W	1.01-04	1.25-05	1.63-05	2.53-05	1.29-05	8.07-05	8.33-06	6.18-06	5.51-06	2.53-05
AG-108	Y	1.69-03	1.15-04	2.20-05	2.83-04	6.62-05	1.82-04	7.92-05	6.22-05	7.44-05	2.83-04
CD-109	D	1.24-05	1.07-05	1.24-05	1.61-04	1.46-03	2.64-04	1.28-05	1.16-05	9.84-06	1.14-04
CD-109	W	5.40-05	3.34-06	1.09-05	5.20-05	4.22-04	7.59-05	3.77-06	3.43-06	2.83-06	3.96-05
CD-109	Y	2.89-04	1.54-06	9.25-06	6.39-05	1.24-04	2.32-05	1.65-06	1.46-06	8.88-07	4.51-05
SN-126	D	5.96-05	4.51-05	7.92-05	8.66-05	5.33-05	4.74-05	2.08-04	4.37-04	4.85-05	8.66-05
SN-126	W	5.59-04	2.22-05	9.66-05	9.95-05	1.89-05	2.27-05	6.25-05	1.23-04	1.81-05	9.95-05
SB-125	D	2.36-06	1.14-06	4.26-06	2.13-06	1.12-06	3.92-06	2.40-06	1.01-05	8.44-07	2.13-06
SB-125	W	8.03-05	2.33-06	1.24-05	1.22-05	1.24-06	3.17-06	1.98-06	3.62-06	1.20-06	1.22-05
I-129	D	7.88-07	4.61-08	4.28-08	2.05-06	4.49-07	4.66-07	6.05-07	5.64-07	5.00-03	1.50-04
CS-134	D	3.38-05	3.26-05	3.71-05	4.55-05	6.77-05	6.99-05	6.16-05	5.89-05	5.19-05	6.55-05
CS-135	D	6.40-07	3.82-08	8.51-08	4.40-06	7.47-06	7.47-06	7.47-06	8.62-06	7.48-06	6.69-06
CS-137	D	1.62-05	1.39-05	1.60-05	3.26-06	5.13-05	5.23-05	4.91-05	5.31-05	4.47-05	4.83-05
EU-152	W	2.13-04	7.36-05	5.55-05	2.21-04	1.37-04	1.28-03	2.93-04	8.88-04	3.05-05	2.21-04
EU-154	W	2.93-04	6.66-05	6.62-05	2.86-04	1.25-04	1.58-03	3.92-04	1.94-03	2.64-05	2.86-04
PB-210	W	6.20-03	1.20-06	4.70-05	3.50-03	3.30-03	3.10-03	2.20-03	2.00-02	6.70-04	2.60-03
AC-227	Y	1.03+00	7.31-05	1.80-03	1.01-01	9.10-02	6.94-01	4.50-01	4.47+00	6.50-03	5.20-01
AC-227	W	8.40-02	3.55-05	2.20-04	1.62-01	2.46-01	1.91+00	1.22+00	1.22+01	1.40-02	6.81-01
TH-228	Y	7.16-01	9.23-05	2.90-03	1.90-02	1.30-03	2.50-03	3.70-02	4.01-01	8.60-04	1.34-01
TH-228	W	1.17-01	2.20-05	7.10-04	2.20-02	2.40-03	1.20-02	2.12-01	2.31+00	2.50-03	1.13-01
TH-229	Y	1.23+00	1.50-04	1.40-03	1.07-01	4.70-03	1.50-02	6.59-01	7.82+00	4.20-03	6.35-01
TH-229	W	1.16-01	8.34-05	3.90-04	1.42-01	4.40-03	2.20-02	1.59+00	1.90+01	4.30-03	7.81-01
RN-222	*	2.34-06	1.55-10	6.86-12	3.39-08	1.37-08	1.35-09	1.68-09	1.03-08	3.02-10	2.83-07
RA-226	W	5.60-02	3.81-06	1.80-04	4.70-03	6.60-04	6.60-04	2.50-03	2.30-02	6.60-04	8.51-03
RA-228	W	4.80-03	9.58-06	6.99-05	2.50-03	5.40-04	7.50-04	1.70-03	1.40-02	5.40-04	1.80-03
TH-230	Y	5.26-01	2.67-06	1.00-04	3.80-02	1.10-03	5.50-03	2.54-01	3.98+00	1.10-03	2.59-01
TH-230	W	5.40-02	2.74-06	9.35-05	5.60-02	2.60-03	1.30-02	6.23-01	9.78+00	2.80-03	3.78-01
TH-232	Y	4.54-01	7.64-06	1.10-04	3.80-02	1.10-03	4.90-03	2.71-01	4.53+00	1.10-03	2.66-01
TH-232	W	4.70-02	8.75-06	9.23-05	5.90-02	2.30-03	1.10-02	6.53-01	1.10+01	2.40-03	4.16-01

[#] Notation: 1.25-07 means 1.25 x 10^{-7}

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TABLE D-7 . DCF2: Fundamental Dose Conversion Factor For Inhalation (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	5.94-01	2.50-05	1.70-04	1.38-01	1.92-01	1.49+00	8.08-01	9.36+00	1.20-02	6.59-01
PA-231 W	5.80-02	3.91-05	1.30-04	2.61-01	4.41-01	3.43+00	1.84+00	2.16+01	2.60-02	1.16+00
U-232 Y	8.87-01	5.58-05	1.10-03	5.30-02	5.80-03	4.50-03	8.80-02	9.54-01	1.80-03	3.24-01
U-232 W	6.70-02	8.54-06	1.30-04	7.60-03	9.70-03	3.70-04	5.70-02	6.29-01	1.90-04	3.50-02
U-232 D	1.10-03	1.86-05	5.12-05	2.00-02	3.20-02	2.90-04	1.72-01	1.91+00	3.00-04	8.02-02
U-233 Y	5.42-01	3.86-06	1.10-04	1.70-02	1.70-03	1.55-05	2.40-04	3.60-03	1.64-05	1.12-01
U-233 W	5.60-02	6.20-06	9.42-05	2.60-03	5.00-03	4.44-05	7.00-04	1.00-02	4.69-05	7.80-03
U-233 D	9.50-04	1.42-05	4.12-05	5.70-03	1.70-02	1.50-04	2.30-03	3.50-02	1.60-04	2.54-03
U-234 Y	5.36-01	3.83-06	1.10-04	1.60-02	1.70-03	1.62-05	2.40-04	3.50-03	1.61-05	1.11-01
U-234 W	5.50-02	6.15-06	9.41-05	2.50-03	4.90-03	4.64-05	6.80-04	1.00-02	4.63-05	7.71-03
U-234 D	9.40-04	1.40-05	4.10-05	5.60-03	1.60-02	1.50-04	2.30-03	3.40-02	1.50-04	2.50-03
U-235 Y	4.84-01	2.48-05	2.50-04	1.50-02	1.50-03	3.16-05	1.90-04	2.90-03	2.07-05	9.97-02
U-235 W	5.00-02	7.84-06	1.10-04	2.30-03	4.40-03	4.12-05	5.30-04	8.50-03	4.28-05	6.93-03
U-235 D	8.50-04	1.42-05	4.36-05	5.10-03	1.50-02	1.30-04	1.80-03	2.80-02	1.40-04	2.15-03
U-236 Y	5.06-01	3.83-06	1.00-04	1.60-02	1.60-03	1.55-05	2.00-04	3.20-03	1.52-05	1.04-01
U-236 W	5.20-02	6.47-06	8.92-05	2.40-03	4.60-03	4.43-05	5.90-04	9.20-03	4.37-05	7.26-03
U-236 D	8.90-04	1.55-05	4.08-05	5.30-03	1.50-02	1.50-04	2.00-03	3.10-02	1.40-04	2.30-03
U-238 Y	4.80-01	8.96-06	2.80-04	1.50-02	1.50-03	1.76-05	2.00-04	2.90-03	1.59-05	9.86-02
U-238 W	4.90-02	6.15-06	1.10-04	2.30-03	4.40-03	4.02-05	5.60-04	8.30-03	4.16-05	6.81-03
U-238 D	8.30-04	1.26-05	3.81-05	5.00-03	1.50-02	1.30-04	1.90-03	2.80-02	1.40-04	2.13-03
NP-237 Y	5.69-01	4.74-05	1.80-04	6.60-02	1.02-01	7.89-01	3.00-01	4.24+00	5.80-03	3.46-01
NP-237 W	5.90-02	7.93-05	1.60-04	1.22-01	2.51-01	1.94+00	7.43-01	1.05+01	1.40-02	5.73-01
PU-236 Y	4.45-01	3.45-06	1.30-04	1.20-02	6.60-03	5.00-02	1.80-02	2.09-01	3.40-04	7.95-02
PU-236 W	6.40-02	2.90-06	1.10-04	1.50-02	3.20-02	2.47-01	8.10-02	9.42-01	1.50-03	6.59-02
PU-238 Y	6.08-01	2.90-06	1.20-04	6.00-02	9.00-02	7.00-01	2.61-01	3.27+00	5.10-03	3.07-01
PU-238 W	6.40-02	2.77-06	1.10-04	1.11-01	2.30-01	1.79+00	6.71-01	8.40+00	1.30-02	4.89-01
PU-239 Y	5.80-01	2.72-06	1.10-04	6.70-02	1.03-01	7.97-01	3.03-01	4.16+00	5.80-03	3.47-01
PU-239 W	6.00-02	2.59-06	1.00-04	1.24-01	2.55-01	1.98+00	7.56-01	1.04+01	1.50-02	5.75-01
PU-240 Y	5.79-01	2.74-06	1.10-04	6.70-02	1.03-01	7.95-01	3.02-01	4.15+00	5.80-03	3.46-01
PU-240 W	6.00-02	2.62-06	1.00-04	1.24-01	2.55-01	1.97+00	7.55-01	1.04+01	1.50-02	6.00-01
PU-241 Y	1.10-03	1.03-07	6.88-07	1.20-03	2.20-03	1.70-02	6.80-03	8.50-02	1.60-04	5.38-03
PU-241 W	1.54-05	1.81-07	6.23-07	2.50-03	4.90-03	3.80-02	1.60-02	1.94-01	3.00-04	1.09-02
PU-242 Y	5.50-01	2.75-06	1.10-04	6.30-02	9.80-02	7.57-01	2.87-01	4.19+00	5.60-03	3.36-01
PU-242 W	5.70-02	2.71-06	9.74-05	1.18-01	2.42-01	1.88+00	7.18-01	1.05+01	1.40-02	5.64-01
PU-244 Y	5.48-01	1.40-04	6.50-04	6.30-02	9.70-02	7.49-01	2.71-01	4.24+00	5.50-03	3.34-01
PU-244 W	5.60-02	1.60-04	3.00-04	1.17-01	2.40-01	1.86+00	6.76-01	1.06+01	1.40-02	5.61-01
AM-241 Y	6.15-01	8.79-06	1.30-04	6.90-02	1.06-01	8.24-01	3.12-01	3.90+00	6.00-03	3.48-01
AM-241 W	6.40-02	1.18-05	1.10-04	1.28-01	2.63-01	2.04+00	7.77-01	9.73+00	1.50-02	5.63-01
AM-243 Y	5.95-01	5.96-05	3.70-04	6.90-02	1.07-01	8.23-01	3.13-01	4.17+00	6.10-03	3.53-01
AM-243 W	6.10-02	8.48-05	1.90-04	1.28-01	2.63-01	2.03+00	7.77-01	1.03+01	1.50-02	5.79-01
CM-242 Y	1.72-01	2.70-06	1.20-04	3.00-03	7.50-04	5.80-03	2.00-03	2.30-02	4.95-05	2.32-02
CM-242 W	5.50-02	2.87-06	1.10-04	3.50-03	6.20-03	4.80-02	1.50-02	1.68-01	3.00-04	1.76-02
CM-243 Y	6.27-01	2.18-05	1.50-04	4.80-02	6.80-02	5.26-01	1.90-01	2.15+00	3.70-03	2.46-01
CM-243 W	6.80-02	3.15-05	1.40-04	8.70-02	1.82-01	1.41+00	5.13-01	5.78+00	1.00-02	3.60-01
CM-244 Y	6.07-01	3.02-06	1.30-04	3.90-02	5.20-02	3.98-01	1.41-01	1.62+00	2.70-03	2.05-01
CM-244 W	6.70-02	2.93-06	1.10-04	6.80-02	1.45-01	1.12+00	3.99-01	4.56+00	7.70-03	2.85-01
CM-248 Y	2.22+00	3.40-03	1.80-03	2.60-01	4.01-01	3.09+00	2.97-01	4.28+00	2.40-02	8.78-01
CM-248 W	2.32-01	5.00-03	3.20-03	4.82-01	9.86-01	7.60+00	7.36-01	1.06+01	5.80-02	1.07+00
CF-252 Y	8.71-01	3.20-04	3.40-04	2.10-02	1.20-02	9.40-02	1.50-02	1.63-01	6.80-04	1.38-01
CF-252 W	1.30-01	2.60-04	4.00-04	2.90-02	6.10-02	4.73-01	7.70-02	8.22-01	3.00-03	8.84-02

Table D-8. DCF3: Fundamental Dose Conversion Factor for External Exposure (Volume Source) (mrem/yr per pCi/m^3)

Nuclide	DCF*	Nuclide	DCF	Nuclide	DCF
H-3	0.00E+00	Cs-135	0.00E+00	U-236	2.62E-10
C-14	0.00E+00	Cs-137	3.39E-06	U-238	6.55E-08
Na-22	1.34E-05	Eu-152	6.20E-06	NP-237	1.10E-06
Cl-36	0.00E+00	Eu-154	6.91E-06	Pu-236	1.46E-10
Fe-55	0.00E+00	Pb-210	1.03E-08	Pu-238	1.09E-10
Co-60	1.55E-05	Rn-222	9.51E-06	Pu-239	0.00E+00
Ni-59	0.00E+00	Ra-226	3.16E-08	Pu-240	1.32E-10
Ni-63	0.00E+00	Ra-228	4.31E-06	Pu-241	1.37E-12
Sr-90	1.92E-13	Ac-227	1.78E-06	Pu-242	1.29E-10
Nb-94	9.50E-06	Th-228	8.62E-06	Pu-244	1.62E-06
Tc-99	1.39E-14	Th-229	1.20E-06	Am-241	7.65E-08
Ru-106	1.11E-06	Th-230	1.07E-09	Am-243	9.41E-07
Ag-108m	8.81E-06	Th-232	0.00E+00	Cm-242	4.81E-10
Cd-109	2.27E-07	Pa-231	2.05E-07	Cm-243	5.55E-07
Sn-126	1.18E-05	U-232	8.32E-10	Cm-244	5.94E-11
Sb-125	2.49E-06	U-233	3.16E-10	Cm-248	0.00E+00
I-129	1.61E-08	U-234	4.89E-10	Cf-252	1.36E-08
Cs-134	9.41E-06	U-235	7.46E-07		

*DCF: fundamental dose conversion factor

TABLE D-9 . DCF4: Fundamental Dose Conversion Factor For External Exposure (Area Source)

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
C-14	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NA-22	D	1.85-04[#]	1.70-04	1.68-04	1.97-04	1.82-04	1.73-04	1.88-04	2.06-04	2.30-04	1.90-04
CL-36	D	2.65-13	6.85-14	1.38-13	3.36-12	4.03-16	3.26-15	8.70-14	4.29-13	1.68-13	1.62-12
CL-36	W	2.65-13	6.85-14	1.38-13	3.36-12	4.03-16	3.26-15	8.70-14	4.29-13	1.68-13	1.62-12
FE-55	W	1.26-09	3.24-10	6.56-10	1.59-08	1.91-12	1.55-11	4.14-10	2.04-09	7.96-10	7.66-09
FE-55	Y	1.26-09	3.24-10	6.56-10	1.59-08	1.91-12	1.55-11	4.14-10	2.04-09	7.96-10	7.66-09
CO-60	W	2.02-04	1.86-04	1.85-04	2.14-04	2.01-04	1.91-04	2.04-04	2.16-04	2.52-04	2.07-04
CO-60	Y	2.02-04	1.86-04	1.85-04	2.14-04	2.01-04	1.91-04	2.04-04	2.16-04	2.52-04	2.07-04
NI-59	D	2.37-09	6.11-10	1.23-09	3.00-08	3.59-12	2.91-11	7.77-10	3.85-09	1.50-09	1.45-08
NI-59	W	2.37-09	6.11-10	1.23-09	3.00-08	3.59-12	2.91-11	7.77-10	3.85-09	1.50-09	1.45-08
NI-63	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NI-63	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NB-94	W	1.40-04	1.30-04	1.27-04	1.49-04	1.38-04	1.31-04	1.42-04	1.55-04	1.76-04	1.44-04
NB-94	Y	1.40-04	1.30-04	1.27-04	1.49-04	1.38-04	1.31-04	1.42-04	1.55-04	1.76-04	1.44-04
TC-99	D	4.93-11	4.30-11	4.04-11	5.52-11	4.56-11	4.44-11	3.62-11	8.63-11	7.26-11	5.08-11
TC-99	W	4.93-11	4.30-11	4.04-11	5.52-11	4.56-11	4.44-11	3.62-11	8.63-11	7.26-11	5.08-11
RU-106	Y	1.84-05	1.70-05	1.66-05	1.97-05	1.80-05	1.72-05	1.88-05	2.11-05	2.30-05	1.90-05
AG-108	D	1.46-04	1.35-04	1.32-04	1.57-04	1.43-04	1.37-04	1.49-04	1.68-04	1.84-04	1.51-04
AG-108	W	1.46-04	1.35-04	1.32-04	1.57-04	1.43-04	1.37-04	1.49-04	1.68-04	1.84-04	1.51-04
AG-108	Y	1.46-04	1.35-04	1.32-04	1.57-04	1.43-04	1.37-04	1.49-04	1.68-04	1.84-04	1.51-04
CD-109	D	7.75-07	5.58-07	5.34-07	1.76-06	9.07-07	4.90-07	3.86-07	1.32-06	1.39-06	1.18-06
CD-109	W	7.75-07	5.58-07	5.34-07	1.76-06	9.07-07	4.90-07	3.86-07	1.32-06	1.39-06	1.18-06
CD-109	Y	7.75-07	5.58-07	5.34-07	1.76-06	9.07-07	4.90-07	3.86-07	1.32-06	1.39-06	1.18-06
SN-126	D	1.81-04	1.67-04	1.63-04	1.93-04	1.76-04	1.68-04	1.82-04	2.09-04	2.27-04	1.86-04
SN-126	W	1.81-04	1.67-04	1.63-04	1.93-04	1.76-04	1.68-04	1.82-04	2.09-04	2.27-04	1.86-04
SB-125	D	3.83-05	3.52-05	3.44-05	4.19-05	3.78-05	3.57-05	3.90-05	4.55-05	4.85-05	4.00-05
SB-125	W	3.83-05	3.52-05	3.44-05	4.19-05	3.78-05	3.57-05	3.90-05	4.55-05	4.85-05	4.00-05
I-129	D	1.07-06	7.81-07	6.70-07	2.02-06	1.77-06	8.44-07	4.03-07	1.87-06	2.17-06	1.47-06
CS-134	D	1.39-04	1.28-04	1.25-04	1.48-04	1.36-04	1.30-04	1.41-04	1.56-04	1.74-04	1.43-04
CS-135	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CS-137	D	5.08-05	4.69-05	4.59-05	5.43-05	4.97-05	5.73-05	5.15-05	5.74-05	6.37-05	5.29-05
EU-152	W	9.73-05	8.92-05	8.81-05	1.04-04	9.66-05	9.10-05	9.69-05	1.11-04	1.24-04	1.00-04
EU-154	W	1.06-04	9.81-05	9.66-05	1.14-04	1.05-04	9.95-05	1.07-04	1.19-04	1.34-04	1.10-04
PB-210	W	1.45-07	1.12-07	9.73-08	2.57-07	1.64-07	1.24-07	6.52-08	2.64-07	2.43-07	1.88-07
AC-227	Y	4.09-05	3.69-05	3.59-05	4.51-05	3.93-05	3.76-05	3.89-05	5.68-05	5.43-05	4.26-05
AC-227	W	4.09-05	3.69-05	3.59-05	4.51-05	3.93-05	3.76-05	3.89-05	5.68-05	5.43-05	4.26-05
TH-228	Y	3.00-04	2.81-04	2.78-04	3.18-04	3.04-04	2.85-04	3.05-04	3.26-04	3.61-04	3.08-04
TH-228	W	3.00-04	2.81-04	2.78-04	3.18-04	3.04-04	2.85-04	3.05-04	3.26-04	3.61-04	3.08-04
TH-229	Y	1.99-04	1.82-04	1.80-04	2.13-04	1.96-04	1.86-04	1.97-04	2.30-04	2.49-04	2.05-04
TH-229	W	1.99-04	1.82-04	1.80-04	2.13-04	1.96-04	1.86-04	1.97-04	2.30-04	2.49-04	2.05-04
RN-222	*	1.48-04	1.36-04	1.35-04	1.57-04	1.46-04	1.39-04	1.50-04	1.65-04	1.65-04	1.51-04
RA-226	W	6.22-07	5.62-07	5.51-07	6.67-07	5.85-07	5.70-07	5.96-07	9.18-07	8.36-07	6.42-07
RA-228	W	8.00-05	7.41-05	7.30-05	8.56-05	7.89-05	7.48-05	8.07-05	8.96-05	1.01-04	8.24-05
TH-230	Y	3.85-08	3.09-08	2.97-08	7.85-08	3.26-08	3.12-08	2.59-08	6.59-08	5.48-08	5.49-08
TH-230	W	3.85-08	3.09-08	2.97-08	7.85-08	3.26-08	3.12-08	2.59-08	6.59-08	5.48-08	5.49-08
TH-232	Y	1.96-08	1.43-08	1.41-08	5.74-08	1.48-08	1.38-08	1.13-08	3.41-08	2.74-08	3.54-08
TH-232	W	1.96-08	1.43-08	1.41-08	5.74-08	1.48-08	1.38-08	1.13-08	3.41-08	2.74-08	3.54-08

[#] Notation: 1.85-04 means 1.85 x 10^{-4}

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TABLE D-9 . DCF4: Fundamental Dose Conversion Factor For External Exposure (Area Source)

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231	Y	2.77-06	2.49-06	2.44-06	3.25-06	2.66-06	2.51-06	2.73-06	3.74-06	3.62-06	2.98-06
PA-231	W	2.77-06	2.49-06	2.44-06	3.25-06	2.66-06	2.51-06	2.73-06	3.74-06	3.62-06	2.98-06
U-232	Y	2.86-08	2.04-08	2.08-08	9.07-08	1.97-08	1.87-08	1.73-08	4.81-08	3.89-08	5.46-08
U-232	W	2.86-08	2.04-08	2.08-08	9.07-08	1.97-08	1.87-08	1.73-08	4.81-08	3.89-08	5.46-08
U-232	D	2.86-08	2.04-08	2.08-08	9.07-08	1.97-08	1.87-08	1.73-08	4.81-08	3.89-08	5.46-08
U-233	Y	2.27-08	1.87-08	1.86-08	4.41-08	1.83-08	1.81-08	1.77-08	3.74-08	3.11-08	3.16-08
U-233	W	2.27-08	1.87-08	1.86-08	4.41-08	1.83-08	1.81-08	1.77-08	3.74-08	3.11-08	3.16-08
U-233	D	2.27-08	1.87-08	1.86-08	4.41-08	1.83-08	1.81-08	1.77-08	3.74-08	3.11-08	3.16-08
U-234	Y	1.74-08	1.11-08	1.19-08	7.07-08	1.00-08	9.33-09	9.15-09	2.95-08	2.31-08	3.98-08
U-234	W	1.74-08	1.11-08	1.19-08	7.07-08	1.00-08	9.33-09	9.15-09	2.95-08	2.31-08	3.98-08
U-234	D	1.74-08	1.11-08	1.19-08	7.07-08	1.00-08	9.33-09	9.15-09	2.95-08	2.31-08	3.98-08
U-235	Y	1.50-05	1.35-05	1.32-05	1.70-05	1.41-05	1.37-05	1.41-05	2.26-05	2.04-05	1.59-05
U-235	W	1.50-05	1.35-05	1.32-05	1.70-05	1.41-05	1.37-05	1.41-05	2.26-05	2.04-05	1.59-05
U-235	D	1.50-05	1.35-05	1.32-05	1.70-05	1.41-05	1.37-05	1.41-05	2.26-05	2.04-05	1.59-05
U-236	Y	1.40-08	8.41-09	9.19-09	6.41-08	6.85-09	6.67-09	6.52-09	2.41-08	1.81-08	3.51-08
U-236	W	1.40-08	8.41-09	9.19-09	6.41-08	6.85-09	6.67-09	6.52-09	2.41-08	1.81-08	3.51-08
U-236	D	1.40-08	8.41-09	9.19-09	6.41-08	6.85-09	6.67-09	6.52-09	2.41-08	1.81-08	3.51-08
U-238	Y	1.73-04	1.60-04	1.57-04	1.85-04	1.70-04	1.62-04	1.73-04	1.98-04	2.19-04	1.78-04
U-238	W	1.73-04	1.60-04	1.57-04	1.85-04	1.70-04	1.62-04	1.73-04	1.98-04	2.19-04	1.78-04
U-238	D	1.73-04	1.60-04	1.57-04	1.85-04	1.70-04	1.62-04	1.73-04	1.98-04	2.19-04	1.78-04
NP-237	Y	2.18-05	1.96-05	1.91-05	2.43-05	2.07-05	1.99-05	2.09-05	3.04-05	2.88-05	2.28-05
NP-237	W	2.18-05	1.96-05	1.91-05	2.43-05	2.07-05	1.99-05	2.09-05	3.04-05	2.88-05	2.28-05
PU-236	Y	1.59-08	8.15-09	9.70-09	8.85-08	4.96-09	4.85-09	6.33-09	2.69-08	2.01-08	4.64-08
PU-236	W	1.59-08	8.15-09	9.70-09	8.85-08	4.96-09	4.85-09	6.33-09	2.69-08	2.01-08	4.64-08
PU-238	Y	1.22-08	5.66-09	7.22-09	7.66-08	2.67-09	2.63-09	4.51-09	2.05-08	1.51-08	3.94-08
PU-238	W	1.22-08	5.66-09	7.22-09	7.66-08	2.67-09	2.63-09	4.51-09	2.05-08	1.51-08	3.94-08
PU-239	Y	8.95-09	6.03-09	6.55-09	3.37-08	4.85-09	4.85-09	5.59-09	1.47-08	1.17-08	1.94-08
PU-239	W	8.95-09	6.03-09	6.55-09	3.37-08	4.85-09	4.85-09	5.59-09	1.47-08	1.17-08	1.94-08
PU-240	Y	1.20-08	5.63-09	7.07-09	7.33-08	2.87-09	2.77-09	4.44-09	2.01-08	1.49-08	3.78-08
PU-240	W	1.20-08	5.63-09	7.07-09	7.33-08	2.87-09	2.77-09	4.44-09	2.01-08	1.49-08	3.78-08
PU-241	Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-241	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-242	Y	1.01-08	4.81-09	6.00-09	6.07-08	2.50-09	2.44-09	3.78-09	1.69-08	1.26-08	3.14-08
PU-242	W	1.01-08	4.81-09	6.00-09	6.07-08	2.50-09	2.44-09	3.78-09	1.69-08	1.26-08	3.14-08
PU-244	Y	2.91-05	2.68-05	2.63-05	3.16-05	2.85-05	2.72-05	2.96-05	3.32-05	3.64-05	3.02-05
PU-244	W	2.91-05	2.68-05	2.63-05	3.16-05	2.85-05	2.72-05	2.96-05	3.32-05	3.64-05	3.02-05
AM-241	Y	2.01-06	1.64-06	1.45-06	2.66-06	1.98-06	1.78-06	1.08-06	3.69-06	3.17-06	2.21-06
AM-241	W	2.01-06	1.64-06	1.45-06	2.66-06	1.98-06	1.78-06	1.08-06	3.69-06	3.17-06	2.21-06
AM-243	Y	2.03-05	1.79-05	1.72-05	2.30-05	1.90-05	1.84-05	1.72-05	3.20-05	2.84-05	2.12-05
AM-243	W	2.03-05	1.79-05	1.72-05	2.30-05	1.90-05	1.84-05	1.72-05	3.20-05	2.84-05	2.12-05
CM-242	Y	1.44-08	6.70-09	8.48-09	8.52-08	2.78-09	2.82-09	5.26-09	2.39-08	1.86-08	4.41-08
CM-242	W	1.44-08	6.70-09	8.48-09	8.52-08	2.78-09	2.82-09	5.26-09	2.39-08	1.86-08	4.41-08
CM-243	Y	1.17-05	1.05-05	1.02-05	1.31-05	1.09-05	1.06-05	1.09-05	1.74-05	1.58-05	1.23-05
CM-243	W	1.17-05	1.05-05	1.02-05	1.31-05	1.09-05	1.06-05	1.09-05	1.74-05	1.58-05	1.23-05
CM-244	Y	1.24-08	5.63-09	7.26-09	7.56-08	2.07-09	2.12-09	4.44-09	2.06-08	1.59-08	3.89-08
CM-244	W	1.24-08	5.63-09	7.26-09	7.56-08	2.07-09	2.12-09	4.44-09	2.06-08	1.59-08	3.89-08
CM-248	Y	9.04-09	4.19-09	5.30-09	5.37-08	1.76-09	1.75-09	3.27-09	1.50-08	1.17-08	2.78-08
CM-248	W	9.04-09	4.19-09	5.30-09	5.37-08	1.76-09	1.75-09	3.27-09	1.50-08	1.17-08	2.78-08
CF-252	Y	1.09-08	5.41-09	6.56-09	5.89-08	2.46-09	2.42-09	4.11-09	1.81-08	1.50-08	3.09-08
CF-252	W	1.09-08	5.41-09	6.56-09	5.89-08	2.46-09	2.42-09	4.11-09	1.81-08	1.50-08	3.09-08

TABLE D-10 . DCF5: Fundamental Dose Conversion Factor For External Exposure (Air Immersion)

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
C-14	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NA-22	D	9.84-03[#]	9.10-03	8.99-03	1.05-02	9.73-03	9.25-03	9.99-03	1.10-02	1.23-02	1.01-02
CL-36	D	2.41-12	6.22-13	1.25-12	3.05-11	3.66-15	2.96-14	7.92-13	3.92-12	1.52-12	1.47-11
CL-36	W	2.41-12	6.22-13	1.25-12	3.05-11	3.66-15	2.96-14	7.92-13	3.92-12	1.52-12	1.47-11
FE-55	W	6.52-09	1.69-09	3.39-09	8.26-08	9.89-12	8.04-11	2.14-09	1.06-08	4.15-09	3.98-08
FE-55	Y	6.52-09	1.69-09	3.39-09	8.26-08	9.89-12	8.04-11	2.14-09	1.06-08	4.15-09	3.98-08
CO-60	W	1.15-02	1.07-02	1.06-02	1.23-02	1.15-02	1.09-02	1.16-02	1.23-02	1.44-02	1.19-02
CO-60	Y	1.15-02	1.07-02	1.06-02	1.23-02	1.15-02	1.09-02	1.16-02	1.23-02	1.44-02	1.19-02
NI-59	D	1.10-08	2.83-09	5.70-09	1.39-07	1.66-11	1.35-10	3.60-09	1.78-08	6.93-09	6.70-08
NI-59	W	1.10-08	2.83-09	5.70-09	1.39-07	1.66-11	1.35-10	3.60-09	1.78-08	6.93-09	6.70-08
NI-63	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NI-63	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NB-94	W	7.22-03	6.70-03	6.56-03	7.70-03	7.11-03	6.78-03	7.33-03	8.04-03	9.11-03	7.43-03
NB-94	Y	7.22-03	6.70-03	6.56-03	7.70-03	7.11-03	6.78-03	7.33-03	8.04-03	9.11-03	7.43-03
TC-99	D	2.09-09	1.82-09	1.71-09	2.34-09	1.93-09	1.89-09	1.53-09	3.65-09	3.07-09	2.15-09
TC-99	W	2.09-09	1.82-09	1.71-09	2.34-09	1.93-09	1.89-09	1.53-09	3.65-09	3.07-09	2.15-09
RU-106	Y	9.22-04	8.52-04	8.33-04	9.85-04	9.04-04	8.63-04	9.41-04	1.06-03	1.15-03	9.50-04
AG-108	D	7.26-03	6.70-03	6.55-03	7.75-03	7.08-03	6.78-03	7.41-03	8.24-03	9.10-03	7.47-03
AG-108	W	7.26-03	6.70-03	6.55-03	7.75-03	7.08-03	6.78-03	7.41-03	8.24-03	9.10-03	7.47-03
AG-108	Y	7.26-03	6.70-03	6.55-03	7.75-03	7.08-03	6.78-03	7.41-03	8.24-03	9.10-03	7.47-03
CD-109	D	3.57-06	2.24-06	2.16-06	1.07-05	4.88-06	1.65-06	1.23-06	5.99-06	7.25-06	6.55-06
CD-109	W	3.57-06	2.24-06	2.16-06	1.07-05	4.88-06	1.65-06	1.23-06	5.99-06	7.25-06	6.55-06
CD-109	Y	3.57-06	2.24-06	2.16-06	1.07-05	4.88-06	1.65-06	1.23-06	5.99-06	7.25-06	6.55-06
SN-126	D	1.96-02	1.82-02	1.77-02	2.10-02	1.93-02	1.84-02	2.00-02	2.25-02	2.47-02	2.02-02
SN-126	W	1.96-02	1.82-02	1.77-02	2.10-02	1.93-02	1.84-02	2.00-02	2.25-02	2.47-02	2.02-02
SB-125	D	1.87-03	1.72-03	1.68-03	2.03-03	1.84-03	1.74-03	1.89-03	2.22-03	2.37-03	1.94-03
SB-125	W	1.87-03	1.72-03	1.68-03	2.03-03	1.84-03	1.74-03	1.89-03	2.22-03	2.37-03	1.94-03
I-129	D	2.06-05	1.51-05	1.30-05	3.81-05	3.39-05	1.65-05	7.81-06	3.63-05	4.15-05	2.80-05
CS-134	D	7.04-03	6.52-03	6.37-03	7.52-03	6.93-03	6.59-03	7.19-03	7.93-03	8.85-03	7.25-03
CS-135	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CS-137	D	2.69-03	2.48-03	2.43-03	2.87-03	2.63-03	2.51-03	2.74-03	3.04-03	3.37-03	2.77-03
EU-152	W	5.19-03	4.78-03	4.70-03	5.52-03	5.15-03	4.85-03	5.19-03	5.81-03	6.56-03	5.32-03
EU-154	W	5.74-03	5.30-03	5.22-03	6.11-03	5.67-03	5.37-03	5.74-03	6.37-03	7.22-03	5.89-03
PB-210	W	4.22-06	3.35-06	2.87-06	5.93-06	5.04-06	3.78-06	1.93-06	7.81-06	7.30-06	4.82-06
AC-227	Y	1.86-03	1.68-03	1.64-03	2.02-03	1.78-03	1.71-03	1.79-03	2.54-03	2.45-03	1.93-03
AC-227	W	1.86-03	1.68-03	1.64-03	2.02-03	1.78-03	1.71-03	1.79-03	2.54-03	2.45-03	1.93-03
TH-228	Y	1.91-02	1.79-02	1.78-02	2.02-02	1.94-02	1.82-02	1.95-02	2.05-02	2.29-02	1.96-02
TH-228	W	1.91-02	1.79-02	1.78-02	2.02-02	1.94-02	1.82-02	1.95-02	2.05-02	2.29-02	1.96-02
TH-229	Y	1.09-02	1.00-02	9.97-03	1.17-02	1.08-02	1.03-02	1.09-02	1.25-02	1.36-02	1.12-02
TH-229	W	1.09-02	1.00-02	9.97-03	1.17-02	1.08-02	1.03-02	1.09-02	1.25-02	1.36-02	1.12-02
RN-222	*	2.15-02	1.99-02	1.97-02	2.28-02	2.14-02	2.02-02	2.17-02	2.36-02	2.65-02	2.20-02
RA-226	W	2.15-02	1.99-02	1.97-02	2.28-02	2.14-02	2.03-02	2.18-02	2.36-02	2.65-02	2.21-02
RA-228	W	4.26-03	3.93-03	3.89-03	4.52-03	4.19-03	4.00-03	4.30-03	4.74-03	5.37-03	4.37-03
TH-230	Y	1.41-06	1.21-06	1.13-06	1.73-06	1.33-06	1.27-06	1.02-06	2.41-06	2.07-06	1.52-06
TH-230	W	1.41-06	1.21-06	1.13-06	1.73-06	1.33-06	1.27-06	1.02-06	2.41-06	2.07-06	1.52-06
TH-232	Y	5.96-07	5.00-07	4.63-07	8.22-07	5.63-07	5.26-07	3.93-07	1.04-06	8.93-07	6.78-07
TH-232	W	5.96-07	5.00-07	4.63-07	8.22-07	5.63-07	5.26-07	3.93-07	1.04-06	8.93-07	6.78-07

[#] Notation: 9.84-03 means 9.84 x 10^{-3}

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TABLE D-10 . DCF5: Fundamental Dose Conversion Factor For External Exposure (Air Immersion)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	1.25-04	1.13-04	1.11-04	1.37-04	1.20-04	1.15-04	1.25-04	1.68-04	1.63-04	1.30-04
PA-231 W	1.25-04	1.13-04	1.11-04	1.37-04	1.20-04	1.15-04	1.25-04	1.68-04	1.63-04	1.30-04
U-232 Y	8.37-07	7.07-07	6.70-07	1.19-06	7.74-07	7.30-07	6.07-07	1.41-06	1.22-06	9.72-07
U-232 W	8.37-07	7.07-07	6.70-07	1.19-06	7.74-07	7.30-07	6.07-07	1.41-06	1.22-06	9.72-07
U-232 D	8.37-07	7.07-07	6.70-07	1.19-06	7.74-07	7.30-07	6.07-07	1.41-06	1.22-06	9.72-07
U-233 Y	8.89-07	7.85-07	7.59-07	1.06-06	8.07-07	7.93-07	7.44-07	1.47-06	1.26-06	9.54-07
U-233 W	8.89-07	7.85-07	7.59-07	1.06-06	8.07-07	7.93-07	7.44-07	1.47-06	1.26-06	9.54-07
U-233 D	8.89-07	7.85-07	7.59-07	1.06-06	8.07-07	7.93-07	7.44-07	1.47-06	1.26-06	9.54-07
U-234 Y	4.11-07	3.37-07	3.19-07	6.78-07	3.74-07	3.46-07	2.76-07	7.11-07	6.07-07	5.16-07
U-234 W	4.11-07	3.37-07	3.19-07	6.78-07	3.74-07	3.46-07	2.76-07	7.11-07	6.07-07	5.16-07
U-234 D	4.11-07	3.37-07	3.19-07	6.78-07	3.74-07	3.46-07	2.76-07	7.11-07	6.07-07	5.16-07
U-235 Y	6.76-04	6.07-04	5.94-04	7.37-04	6.33-04	6.16-04	6.38-04	1.01-03	9.16-04	7.01-04
U-235 W	6.76-04	6.07-04	5.94-04	7.37-04	6.33-04	6.16-04	6.38-04	1.01-03	9.16-04	7.01-04
U-235 D	6.76-04	6.07-04	5.94-04	7.37-04	6.33-04	6.16-04	6.38-04	1.01-03	9.16-04	7.01-04
U-236 Y	2.99-07	2.39-07	2.21-07	5.33-07	2.61-07	2.48-07	1.72-07	5.41-07	4.48-07	3.90-07
U-236 W	2.99-07	2.39-07	2.21-07	5.33-07	2.61-07	2.48-07	1.72-07	5.41-07	4.48-07	3.90-07
U-236 D	2.99-07	2.39-07	2.21-07	5.33-07	2.61-07	2.48-07	1.72-07	5.41-07	4.48-07	3.90-07
U-238 Y	9.05-03	8.33-03	8.18-03	9.65-03	8.90-03	8.45-03	9.07-03	1.03-02	1.14-02	9.30-03
U-238 W	9.05-03	8.33-03	8.18-03	9.65-03	8.90-03	8.45-03	9.07-03	1.03-02	1.14-02	9.30-03
U-238 D	9.05-03	8.33-03	8.18-03	9.65-03	8.90-03	8.45-03	9.07-03	1.03-02	1.14-02	9.30-03
NP-237 Y	9.84-04	8.91-04	8.69-04	1.07-03	9.39-04	9.05-04	9.52-04	1.36-03	1.30-03	1.02-03
NP-237 W	9.84-04	8.91-04	8.69-04	1.07-03	9.39-04	9.05-04	9.52-04	1.36-03	1.30-03	1.02-03
PU-236 Y	2.10-07	1.53-07	1.46-07	5.33-07	1.64-07	1.51-07	1.06-07	3.78-07	3.10-07	3.41-07
PU-236 W	2.10-07	1.53-07	1.46-07	5.33-07	1.64-07	1.51-07	1.06-07	3.78-07	3.10-07	3.41-07
PU-238 Y	1.18-07	7.85-08	7.78-08	3.96-07	8.04-08	7.15-08	5.30-08	2.09-07	1.71-07	2.33-07
PU-238 W	1.18-07	7.85-08	7.78-08	3.96-07	8.04-08	7.15-08	5.30-08	2.09-07	1.71-07	2.33-07
PU-239 Y	2.42-07	2.07-07	2.03-07	3.65-07	2.09-07	2.05-07	1.95-07	4.00-07	3.39-07	2.92-07
PU-239 W	2.42-07	2.07-07	2.03-07	3.65-07	2.09-07	2.05-07	1.95-07	4.00-07	3.39-07	2.92-07
PU-240 Y	1.21-07	8.15-08	8.00-08	3.89-07	8.63-08	7.59-08	5.44-08	2.15-07	1.78-07	2.31-07
PU-240 W	1.21-07	8.15-08	8.00-08	3.89-07	8.63-08	7.59-08	5.44-08	2.15-07	1.78-07	2.31-07
PU-241 Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-241 W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-242 Y	1.07-07	7.41-08	7.22-08	3.30-07	7.78-08	7.00-08	5.00-08	1.92-07	1.58-07	1.99-07
PU-242 W	1.07-07	7.41-08	7.22-08	3.30-07	7.78-08	7.00-08	5.00-08	1.92-07	1.58-07	1.99-07
PU-244 Y	1.48-03	1.37-03	1.34-03	1.58-03	1.45-03	1.39-03	1.51-03	1.68-03	1.85-03	1.52-03
PU-244 W	1.48-03	1.37-03	1.34-03	1.58-03	1.45-03	1.39-03	1.51-03	1.68-03	1.85-03	1.52-03
AM-241 Y	6.93-05	5.70-05	5.04-05	8.41-05	6.93-05	6.22-05	3.74-05	1.27-04	1.10-04	7.29-05
AM-241 W	6.93-05	5.70-05	5.04-05	8.41-05	6.93-05	6.22-05	3.74-05	1.27-04	1.10-04	7.29-05
AM-243 Y	8.86-04	7.86-04	7.53-04	9.77-04	8.31-04	8.04-04	7.63-04	1.38-03	1.23-03	9.16-04
AM-243 W	8.86-04	7.86-04	7.53-04	9.77-04	8.31-04	8.04-04	7.63-04	1.38-03	1.23-03	9.16-04
CM-242 Y	1.26-07	8.19-08	8.33-08	4.48-07	7.59-08	7.00-08	5.74-08	2.22-07	1.82-07	2.59-07
CM-242 W	1.26-07	8.19-08	8.33-08	4.48-07	7.59-08	7.00-08	5.74-08	2.22-07	1.82-07	2.59-07
CM-243 Y	5.22-04	4.70-04	4.59-04	5.70-04	4.93-04	4.78-04	4.89-04	7.74-04	7.07-04	5.42-04
CM-243 W	5.22-04	4.70-04	4.59-04	5.70-04	4.93-04	4.78-04	4.89-04	7.74-04	7.07-04	5.42-04
CM-244 Y	9.70-08	5.96-08	6.26-08	3.81-07	5.26-08	4.78-08	4.19-08	1.69-07	1.39-07	2.15-07
CM-244 W	9.70-08	5.96-08	6.26-08	3.81-07	5.26-08	4.78-08	4.19-08	1.69-07	1.39-07	2.15-07
CM-248 Y	7.81-08	5.00-08	5.07-08	2.81-07	4.70-08	4.26-08	3.43-08	1.37-07	1.13-07	1.62-07
CM-248 W	7.81-08	5.00-08	5.07-08	2.81-07	4.70-08	4.26-08	3.43-08	1.37-07	1.13-07	1.62-07
CF-252 Y	1.10-07	7.41-08	7.48-08	3.47-07	6.70-08	6.37-08	5.41-08	1.91-07	1.60-07	2.07-07
CF-252 W	1.10-07	7.41-08	7.48-08	3.47-07	6.70-08	6.37-08	5.41-08	1.91-07	1.60-07	2.07-07

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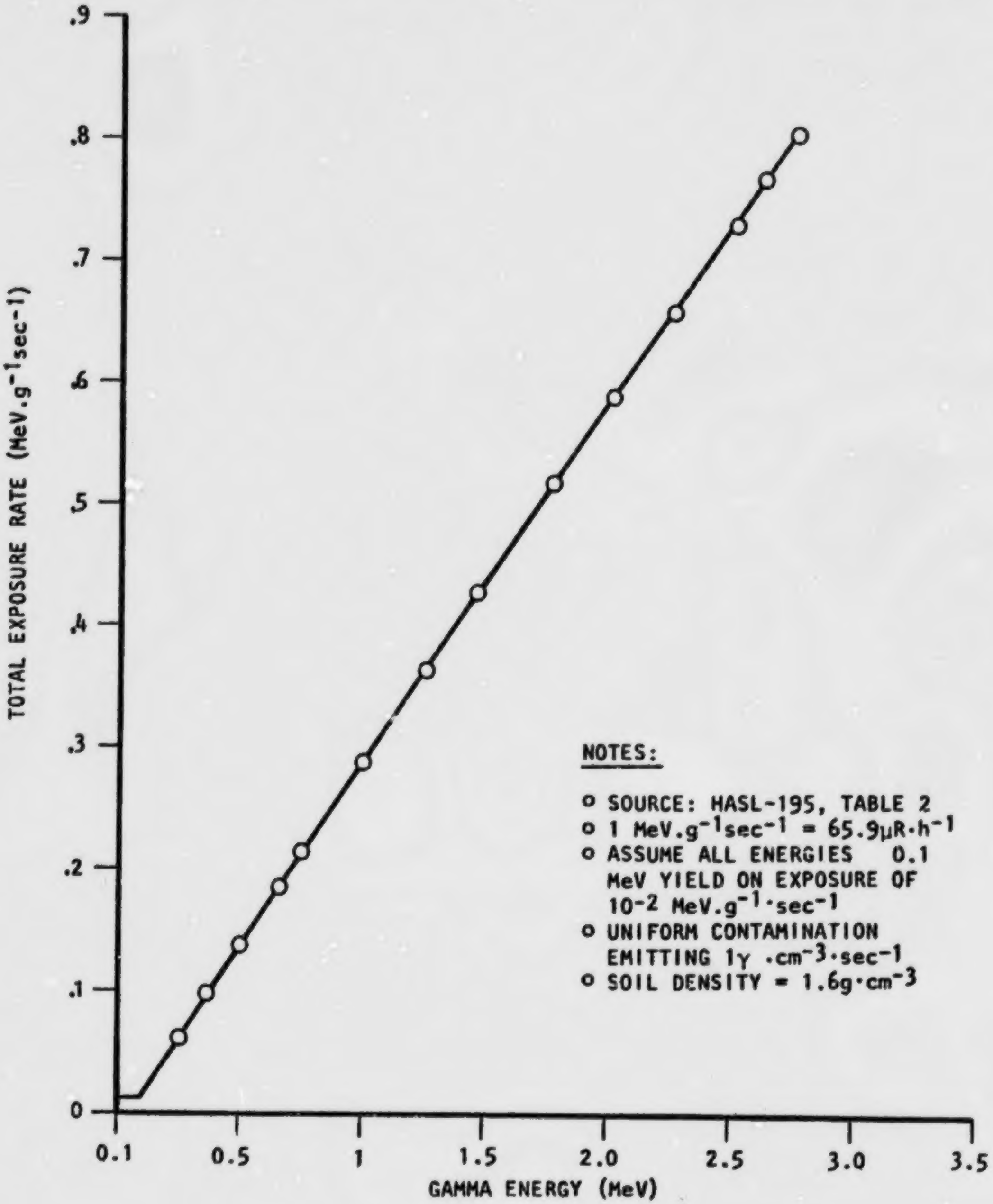

NOTES:

- o SOURCE: HASL-195, TABLE 2
- o 1 MeV.g^{-1}sec^{-1} = 65.9μR$\cdot$h^{-1}
- o ASSUME ALL ENERGIES 0.1 MeV YIELD ON EXPOSURE OF 10^{-2} MeV.g$^{-1}\cdot$sec^{-1}
- o UNIFORM CONTAMINATION EMITTING 1γ $\cdot$cm$^{-3}\cdot$sec^{-1}
- o SOIL DENSITY = 1.6g$\cdot$cm^{-3}

TOTAL EXPOSURE RATE VS. GAMMA ENERGY
AT ONE METER ABOVE BULK CONTAMINATED SOIL

Figure D.1

Exposure rates, E, for the radionuclides of interest in this report were then calculated from the expression:

$$E = K \Sigma f_i E_T \qquad (D-10)$$

where f_i is the fraction of gamma photons of energy T per disintegration, E_T is the exposure rate factor obtained from Figure D.1 for energy T, and K is a proportionality constant which converts the exposure rate factors in HASL-195 to units of dose equivalent (mrem/year per Ci/m^3). In this case, K = 65.9 μR/hr per MeV/g-sec, and it is assumed that one Roentgen equals one rem. Gamma energies and fractional data for the various radionuclides were obtained from reference 10. Calculations were performed with the aid of a small computer program; a listing for this program is provided in Volume 2 of this report.

The resultant annual external gamma dose conversion factors resulting from volume contaminated soil (DCF3) are presented in Table D-8. These DCFs are for whole body only. DCFs for other organs are approximated using scaling factors calculated from the ratios of the organ-dependent DCFs for DCF4. (DCF4 is discussed below.)

Other direct exposure DCFs. The two remaining fundamental DCFs used in this report are the external exposure dose conversion factors resulting from direct photon and electron radiation emanating from a contaminated surface, and from immersion in uniformly contaminated air. These DCFs are denoted by DCF4 and DCF5, respectively.

In the past, the electron component (beta radiation) of the exposure was frequently neglected in comparison to the photon component (gamma radiation) of the exposure due to the comparative penetration capabilities of these radiations. This is the case for DCF3, since a few millimeters of soil is sufficient to stop most of the electron radiation from the radionuclides considered in this work. It is more accurate to include the electron component when the exposure is due to a surface contamination or to immersion in contaminated air.

These DCFs are presented in Tables D-9 and D-10 in units of mrem per year for a unit concentration of a radionuclide at the biota access location--i.e., pCi/m^2 for DCF4 and pCi/m^3 for DCF5. All DCFs have been obtained from reference 11. For each case, DCFs for photons and electrons have been calculated for tissue-equivalent material at the body surface of an exposed individual. For internal body organs, only photons have been considered (Ref. 11).

D.2.3 Translocation Factors

The remaining parameters in the equations listed in Table D-3 may be termed translocation parameters (also referred to as uptake factors or pathway parameters). Some of these are radionuclide-specific and are used to track the transfer of radionuclides through food products. Five sets of these transfer factors as obtained from the literature are listed in Tables D-11 through D-14. A number of literature sources have been considered, and a comparative compilation of the factors obtained from the literature is presented in the tables, along with the actual factors used in this report. These five sets of transfer factors are summarized below:

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Symbol	Definition	Units	Table
f_1	Soil-to-Plant Transfer Factor (pCi/kg in fresh vegetation per pCi/kg in soil)	Dimensionless	D-11
f_4	Feed or Water-to-Meat Transfer Factor (pCi/kg in meat per pCi/day ingested by beef cattle)	day/kg	D-12
f_6	Feed or Water-to-Animal Product (Milk) Transfer Factor (pCi/ℓ in milk per pCi/day ingested by cow)	day/ℓ	D-13
f_{12}	Water-to-Fish Transfer Factor (pCi/kg of fresh fish per pCi/ℓ of water concentration)	ℓ/kg	D-14
f_{12}^P	Water-to-Freshwater Seafood Transfer Factor (pCi/kg of fresh seafood per pCi/ℓ of water concentration)	ℓ/kg	D-14

In selecting a particular transfer factor for use in this report, more recent references from the literature have generally been utilized. In doing so, the intent was to try to provide a reasonable yet conservative estimate of radionuclide transfer. Examination of the tables reveals for a number of elements a considerable spread in transfer factor values depending upon the reference cited. This is not suprising when one considers that the cited transfer factors generally represent averages of a number of measurements involving different types of plant and animal food products. For example, Table D-11 lists factors for radionuclide uptake by means of root uptake from soil to plants used for food products. The transfer factors represented in the various citations were selected by the citation authors by considering uptake factors measured for several different types of plants, and these uptake factors can vary significantly depending upon the plant, environmental conditions, and so forth. The citations are furthermore believed to give generally conservative transfer factors. They are often meant as conservative "default" values to be used in the absence of site-specific information on crop production, environmental characteristics, and so forth.

For many of the transfer factors the inherent uncertainty makes little difference for the kinds of impact scenarios considered in this report. This is because impacts are dominated for many radionuclides by exposures from pathways other than food pathways. This is the case for impacts associated with Co-60 and Cs-137, for example, which are dominated by direct external radiation pathways, or impacts associated with Pu-239, which are dominated by inhalation pathways. For some radionuclides, however, food ingestion pathways are more significant, and so for these radionuclides selection of the transfer factors becomes more critical. Radionuclides which fall into this category include (among others) carbon-14, strontium-90, technetium-99, and radium-226. For carbon-14, for example, reference 16 indicates that the currently used soil-to-plant transfer factor may be as much as a factor of 10 too large. As another example, Table D-15 presents a comparison of ranges of transfer factors for technetium-99 (Ref. 24). Additional work is recommended in determining appropriate transfer factors for these and other radionuclides of concern.

Other parameters used in the calculations are listed in Tables D-16 through D-18.

D.2.4 Lists of Pathway Dose Conversion Factors

The pathway dose conversion factors used in this report vary depending upon the specific region considered. As an example, those for the Southeast region (IR=2) are listed in Tables D-19 through D-26.

These PDCFs are given in terms of individual radionuclides and solubility classes, and also for nine organs of the body plus an effective whole body equivalent as calculated using ICRP-26 and ICRP-30 methodologies. All PDCFs, except PDCF-8, are given in units of mrem/yr per Ci/m^3. PDCF-8 is given in units of mrem/yr per Ci/m^2.

Table D-11. f_1: Soil-to-Plant Transfer Factors (dimensionless)

Element	Ref 1	Ref 2	Ref 12	Ref 13	Ref 14	Ref 15
Hydrogen	4.8E+0(a)	4.8E+0				
Carbon	5.5E+0(b)	5.5E+0				
Sodium	5.2E-2	5.2E-2			4.6E-3	
Chlorine	5.0E+0					
Iron	6.6E-4	6.6E-4			4.2E-4	
Cobalt	9.4E-3	9.4E-3			1.5E-2	
Nickel	1.9E-2	1.9E-2			2.1E-2	
Strontium	1.7E-2	1.7E-2		2.9E-1	7.5E-2	
Niobium	9.4E-3	9.4E-3				
Technetium	2.5E-1	2.5E-1		1.1E+0		
Ruthenium	5.0E-1	5.0E-2			1.4E-3	
Silver	1.5E-1	1.5E-1				
Cadmium	3.0E-1					
Tin	2.5E-3					
Antimony	1.1E-2					
Iodine	2.0E-2	2.0E-2		5.5E-2	4.5E-3	
Cesium	1.0E-2	1.0E-2		9.3E-3	5.0E-3	
Europium	2.5E-3					
Lead	6.8E-2		4.0E-3	3.9E-3		
Radon	3.5E+0					
Radium	3.1E-4		1.4E-2	6.2E-2		
Actinium	2.5E-3			2.5E-3		
Thorium	4.2E-3		4.2E-3	3.5E-4		
Protactinium	2.5E-3			2.5E-3		
Uranium	2.5E-3		2.5E-3	2.9E-4		
Neptunium	2.5E-3	2.5E-3			6.0E-2	
Plutonium	2.5E-4			2.0E-4		5.6E-4
Americium	2.5E-4					5.6E-3
Curium	2.5E-3					
Californium	2.5E-3					

(a) Values selected for use in this report have been underlined.

(b) For carbon-14, Reference 16 suggests that a transfer factor equal to 5.5E-1 may be more appropriate.

Table D-12. f_4: Feed- and Water-to-Meat Transfer Factors (day/kg)

Element	Ref 1	Ref 2	Ref 12	Ref 13	Ref 17	Ref 15
Hydrogen	1.2E-2(a)	1.2E-2				
Carbon	3.1E-2	3.1E-2				
Sodium	3.0E-2	3.0E-2			8.3E-2	
Chlorine	8.0E-2					
Iron	4.0E-2	4.0E-2			1.9E-2	
Cobalt	1.3E-2	1.3E-2			9.7E-3	
Nickel	5.3E-3	5.3E-3			2.0E-3	
Strontium	6.0E-4	6.0E-4		3.0E-4	5.9E-4	
Niobium	2.8E-1	2.8E-1			2.5E-1	
Technetium	4.0E-1	4.0E-1		8.7E-3		
Ruthenium	4.0E-1	4.0E-1				
Silver	1.7E-2	1.7E-2			1.9E-3	
Cadmium	5.3E-4					
Tin	8.0E-2					
Antimony	4.0E-3					
Iodine	2.9E-3	2.9E-3		7.0E-3		
Cesium	4.0E-3	4.0E-3		1.4E-2		
Europium	4.8E-3					
Lead	2.9E-4		7.1E-4	9.1E-4		
Radon	2.0E-2					
Radium	3.4E-2		5.1E-4	5.0E-4		
Actinium	6.0E-2			1.6E-6		
Thorium	2.0E-4		2.0E-4	1.6E-6		
Protactinium	8.0E-2			1.6E-6		
Uranium	2.4E-4		3.4E-4	1.6E-6		
Neptunium	2.0E-4	2.0E-4				
Plutonium	1.4E-5			4.1E-7		3.9E-4
Americium	2.0E-4					3.9E-3
Curium	2.0E-4					
Californium	2.0E-4					

(a) Values selected for use in this report have been underlined.

Table D-13. f_6: Feed- and Water-to-Milk Transfer Factors (day/ℓ)

Element	Ref 1	Ref 2	Ref 12	Ref 13	Ref 18	Ref 19
Hydrogen	1.0E-2[a]	1.0E-2			1.4E-2	1.4E-2
Carbon	1.2E-2	1.2E-2			1.5E-2	1.5E-2
Sodium	4.0E-2	4.0E-2			3.5E-2	3.5E-2
Chlorine	5.0E-2				1.7E-2	1.7E-2
Iron	1.2E-3	1.3E-3			2.7E-4	5.9E-5
Cobalt	1.0E-3	1.0E-3			1.8E-3	2.0E-3
Nickel	6.7E-3	6.7E-3			1.0E-3	1.0E-3
Strontium	8.0E-4	8.0E-4		2.4E-3	1.4E-3	1.4E-3
Niobium	2.5E-3	2.5E-3				2.0E-2
Technetium	2.5E-2	2.5E-2		9.9E-3		9.9E-3
Ruthenium	1.0E-6	1.0E-6			6.1E-7	6.1E-7
Silver	5.0E-2	5.0E-2			1.9E-2	1.0E-2
Cadmium	1.2E-4				1.0E-3	2.0E-3
Tin	2.5E-3				1.2E-3	1.2E-3
Antimony	1.5E-3				1.1E-4	2.0E-5
Iodine	6.0E-3	6.0E-3		1.0E-2	9.9E-3	9.9E-3
Cesium	1.2E-2	1.2E-2		5.6E-3	7.1E-3	7.1E-3
Europium	5.0E-6					2.0E-5
Lead	6.2E-4		1.2E-4	9.9E-5	2.6E-4	2.6E-4
Radon	2.0E-2					2.0E-2
Radium	8.0E-3		5.9E-4	5.9E-4	4.5E-4	4.5E-4
Actinium	5.0E-6			2.0E-5		2.0E-5
Thorium	5.0E-6		5.0E-6	5.0E-6		5.0E-6
Protactinium	5.0E-6			5.0E-6		5.0E-6
Uranium	5.0E-4		6.1E-4	1.2E-4	6.1E-4	6.1E-4
Neptunium	5.0E-6	5.0E-6				5.0E-6
Plutonium	2.0E-6			4.5E-8	1.0E-7	1.0E-7
Americium	5.0E-6				4.1E-7	2.0E-5
Curium	5.0E-6					2.0E-5
Californium	5.0E-6					2.0E-5

(a) Values selected for use in this report have been underlined.

Table D-14. Water-to-Seafood Transfer Factors (ℓ/kg)

Element	Freshwater Fish (f_{12}) Ref 1	Freshwater Fish (f_{12}) Ref 2	Freshwater Fish (f_{12}) Ref 15	Freshwater Seafood (f_{12}^P) Ref 1	Freshwater Seafood (f_{12}^P) Ref 2
Hydrogen	9.0E-1(a)	9.0E-1		9.0E-1	9.0E-1
Carbon	4.6E+3	4.6E+3		9.1E+3	9.1E+3
Sodium	1.0E+2	1.0E+2		2.0E+2	2.0E+2
Chlorine	5.0E+1			1.0E+2	
Iron	1.0E+2	1.0E+2		3.2E+3	3.2E+3
Cobalt	5.0E+1	5.0E+1		2.0E+2	2.0E+2
Nickel	1.0E+2	1.0E+2		1.0E+2	1.0E+2
Strontium	3.0E+1	3.0E+1		1.0E+2	1.0E+2
Niobium	3.0E+4	3.0E+4		1.0E+2	1.0E+2
Technetium	1.5E+1	1.5E+1		5.0E+0	5.0E+0
Ruthenium	1.0E+1	1.0E+1		3.0E+2	3.0E+2
Silver	2.3E+0			7.7E+2	
Cadmium	2.0E+2			2.0E+3	
Tin	3.0E+3			1.0E+3	
Antimony	1.0E+0			1.0E+1	
Iodine	1.5E+1	1.5E+1		5.0E+0	5.0E+0
Cesium	2.0E+3	2.0E+3		1.0E+2	1.0E+3
Europium	2.5E+1			1.0E+3	
Lead	1.0E+2			1.0E+2	
Radon	1.0E+0			1.0E+0	
Radium	5.0E+1			2.5E+2	
Actinium	2.5E+1			1.0E+3	
Thorium	3.0E+1			5.0E+2	
Protactinium	1.1E+1			1.1E+2	
Uranium	2.0E+0			6.0E+1	
Neptunium	1.0E+1	1.0E+1		4.0E+2	4.0E+2
Plutonium	3.5E+0		2.5E+1	1.0E+2	
Americium	2.5E+1		2.5E+2	1.0E+3	
Curium	2.5E+1			1.0E+3	
Californium	2.5E+1			1.0E+3	

(a) Values selected for use in this report have been underlined.

Table D-15. Comparison of Ranges of Selected Transfer Factors for Tc-99

Transfer Factor	Range* Min.	E.V.**	Max.	Transfer To:	Value Used in This Report
Soil to plant	1	2	20	vegetables	1.1
To meat (day/kg)	1E-7	1E-5	1E-4	beef	8.7E-3
	2E-6	2E-4	2E-3	pork	
	6E-4	6E-2	6E-1	chicken	
	8E-4	8E-2	8E-1	eggs (chicken)	
To milk (day/ℓ)	1E-6	1E-4	1E-3	cows	9.9E-3
	4E-5	2E-3	1E-2	goats	
To fresh water organisms (day/ℓ)	10	30	100	fish	15
	5	100	1,000	invertebrates	5

*Source: Table 15 of reference 24.
**E.V.: expected value.

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Table D-16. Radionuclide Independent Parameters Used in Calculations

Symbol	Definition	Value	Reference
CY	Crop Yield per unit area	1 Kg/m^2	20
D	Soil Density	1600 Kg/m^3	20
f_2	Consumption of plants by man	190 Kg/year	2
f_5	Consumption of animals by man	95 Kg/year	2
f_7	Consumption of milk by man	0.3 ℓ/day	2
f_8	Consumption of water by beef cattle	50 ℓ/day	2
f_8^P	Consumption of water by milk cows	60 ℓ/day	2
f_{11}	Consumption of water by man	370 ℓ/year	2
f_{13}	Consumption of fish by man	6.9 Kg/year	2
f_{13}^P	Consumption of seafood by man	1.0 Kg/year	2
f_{14}	Resuspension factor	8.5E-9 m^{-1}	21
f_{15}	Inhalation rate of man	8.0E+3 m^3/year	22
f_{18}	Areal mass available for resuspension (top 1 cm of soil)	16 Kg/m^2	20
R	The fraction of initial activity deposited as fallout or contaminated water that is retained by foliage	0.25	20
S_1	Fraction of activity deposited on foliage removed per unit time by weathering mechanisms	4.83E-2 day^{-1}	20
S_2	Fraction of activity deposited in the root zone removed per unit time	7.6E-04 day^{-1}	20
V	Settling velocity		
	V_1: elements other than iodine	8.0E-4 m/sec	20
	V_2: iodine	1.0E-2 m/sec	20
Z	Mass of soil in root zone	240 Kg/m^2	2

Table 17. Region-Specific Parameters for f_3 (Rate of Plant Consumption by Animals) and RI (Irrigation Rate)

Region	f_3 (Kg/day)	RI (m^3/m^2-day)
Northeast	25	6.8E-4
Southeast	36	6.8E-4
Midwest	22	2.3E-3
Southwest	36	2.7E-3

Source: references 25 and 26.

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Table D-18. Intermediate Parameters Used in Calculations

Symbol	Transfer Factor	Description
P_1	f_1*f_2	Soil-Plant-Man
P_2	$f_1*f_3*f_4*f_5$	Soil-Plant-Animal-Man
P_3	$f_1*f_3*f_6*f_7*365$	Soil-Plant-Animal-Product-Man
PT	$P_1+P_2+P_3$	Total Soil-to-Plant-to-Man (kg/yr)
P_1P	f_2	Plant-Man
P_2P	$f_3*f_4*f_5$	Plant-Animal-Man
P_3P	$f_3*f_6*f_7*365$	Plant-Animal-Product-Man
PTP	$P_1P+P_2P+P_3P$	Total Plant-to-Man (kg/yr)
F_1	$f_8*f_4*f_5$	Water-Animal-Man
F_2	$f_8P*f_6*f_7*365$	Water-Animal-Product-Man
F_3	f_{11}	Water-Man
FT	$F_1+F_2+F_3$	Total Water-to-Man (ℓ/yr)
F_{12_1}	$f_{12}*f_{13}$	Water-Fish-Man
F_{12_2}	$f_{12}P*f_{13}P$	Water-Seafood-Man
F_{12}	$F_{12_1}+F_{12_2}$	Total Seafood-to-Man (ℓ/yr)
D_1	$86400*V/(S_2*Z)$	Soil Deposition by Fallout (m^3/kg)
D_2	$86400*R*V/S_1$	Foliar Deposition by Fallout (m)
W_1	$RI/(S_2*Z)$	Soil Deposition by Irrigation (m^3/kg)
W_2	$R*RI/S_1$	Foliar Deposition by Irrigation (m)
C	1.0E+12	Conversion Factor (pCi/Ci)
	86400	seconds/day
	365	days/year

TABLE D-19 . PDCF-1: PDCF for Chronic Airborne Pathways

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	1.00+09[#]	1.00+09	1.06+09	1.00+09	1.03+09	9.92+08	9.92+08	7.88+08	9.92+08	1.01+09
C-14	*	4.94+07	5.88+07	5.78+07	1.13+08	6.34+07	7.10+07	1.94+08	4.06+08	5.18+07	8.72+07
NA-22	D	1.20+12	1.08+12	1.08+12	1.27+12	1.19+12	1.13+12	1.25+12	1.38+12	1.46+12	1.25+12
CL-36	D	3.94+10	1.63+10	1.49+10	1.93+10	1.49+10	1.49+10	1.49+10	1.49+10	1.49+10	1.79+10
CL-36	W	1.35+12	1.97+10	1.49+10	2.50+11	1.49+10	1.49+10	1.49+10	1.49+10	1.49+10	1.75+11
FE-55	W	2.56+10	8.52+07	2.92+09	5.13+09	8.72+09	1.66+10	8.64+09	5.93+09	8.57+09	1.26+10
FE-55	Y	2.01+11	1.04+08	3.13+09	4.97+09	4.04+09	7.66+09	4.03+09	2.76+09	3.97+09	3.17+10
CO-60	W	2.27+12	1.28+12	1.35+12	1.44+12	1.37+12	1.39+12	1.38+12	1.43+12	1.65+12	1.55+12
CO-60	Y	1.16+13	1.93+12	1.35+12	1.96+12	1.69+12	2.12+12	1.76+12	1.72+12	2.01+12	3.19+12
NI-59	D	1.07+10	1.02+10	1.14+10	1.07+10	1.02+10	1.02+10	1.05+10	1.04+10	1.11+10	1.06+10
NI-59	W	3.55+10	3.14+09	6.64+09	7.53+09	3.08+09	3.08+09	3.14+09	3.14+09	3.39+09	7.43+09
NI-63	D	2.58+10	2.44+10	2.82+10	2.48+10	2.44+10	2.43+10	2.43+10	2.43+10	2.43+10	2.48+10
NI-63	W	9.12+10	7.61+09	1.99+10	1.84+10	7.34+09	7.31+09	7.31+09	7.31+09	7.31+09	1.84+10
SR-90	D	7.91+10	1.58+09	1.13+11	1.92+12	1.17+11	1.17+11	8.80+12	1.76+13	1.17+11	1.68+12
SR-90	Y	6.80+13	2.29+11	7.12+12	1.20+12	2.92+10	1.52+11	9.60+11	1.84+12	2.92+10	1.16+13
NB-94	W	2.09+12	8.95+11	1.02+12	1.19+12	1.04+12	9.07+11	1.05+12	1.21+12	1.15+12	1.16+12
NB-94	Y	2.30+13	1.73+12	1.05+12	4.22+12	1.41+12	1.96+12	1.53+12	1.52+12	1.73+12	4.19+12
TC-99	D	7.70+09	3.92+09	4.07+09	1.35+09	2.78+09	3.82+09	1.95+09	2.49+09	8.56+10	5.66+09
TC-99	W	4.18+11	4.56+09	1.33+10	7.10+09	2.46+09	3.37+09	1.72+09	2.20+09	7.57+10	5.80+10
RU-106	Y	3.05+13	1.59+11	1.22+12	6.14+11	1.81+11	1.96+11	1.89+11	2.08+11	2.13+11	4.29+12
AG-108	D	1.06+12	9.61+11	8.86+11	1.19+12	1.11+12	2.76+12	9.96+11	1.09+12	1.15+12	1.16+12
AG-108	W	1.69+12	9.20+11	9.32+11	1.16+12	9.72+11	1.48+12	9.72+11	1.07+12	1.16+12	1.12+12
AG-108	Y	1.44+13	1.74+12	9.78+11	3.22+12	1.40+12	2.29+12	1.54+12	1.52+12	1.71+12	3.18+12
CD-109	D	1.04+11	8.90+10	1.02+11	1.30+12	1.17+13	2.12+12	1.05+11	1.01+11	8.71+10	9.19+11
CD-109	W	4.37+11	3.01+10	9.04+10	4.27+11	3.38+12	6.10+11	3.25+10	3.54+10	3.10+10	3.24+11
CD-109	Y	2.32+12	1.57+10	7.72+10	5.22+11	9.98+11	1.89+11	1.55+10	1.96+10	1.55+10	3.68+11
SN-126	D	1.59+12	1.38+12	1.63+12	1.88+12	1.51+12	1.41+12	2.78+12	4.78+12	1.78+12	1.83+12
SN-126	W	5.58+12	1.20+12	1.77+12	1.98+12	1.23+12	1.21+12	1.62+12	2.27+12	1.54+12	1.94+12
SB-125	D	2.51+11	2.23+11	2.43+11	2.71+11	2.38+11	2.48+11	2.56+11	3.57+11	3.01+11	2.60+11
SB-125	W	8.75+11	2.32+11	3.08+11	3.52+11	2.39+11	2.42+11	2.53+11	3.05+11	3.04+11	3.40+11
I-129	D	8.69+10	5.92+10	5.08+10	1.69+11	1.37+11	6.73+10	3.52+10	1.45+11	4.02+13	1.31+12
CS-134	D	1.11+12	1.04+12	1.06+12	1.26+12	1.37+12	1.35+12	1.35+12	1.42+12	1.47+12	1.39+12
CS-135	D	5.12+09	3.06+08	6.81+08	3.52+10	5.98+10	5.98+10	5.98+10	6.90+10	5.98+10	5.35+10
CS-137	D	4.38+11	3.96+11	4.07+11	3.56+11	7.12+11	7.66+11	7.06+11	7.74+11	7.45+11	7.08+11
EU-152	W	2.30+12	1.13+12	9.79+11	2.40+12	1.68+12	1.08+13	2.93+12	7.78+12	9.97+11	2.38+12
EU-154	W	2.99+12	1.13+12	1.12+12	2.98+12	1.64+12	1.32+13	3.79+12	1.62+13	1.03+12	2.96+12
PB-210	W	4.96+13	1.03+10	3.77+11	2.80+13	2.64+13	2.48+13	1.76+13	1.60+14	5.36+12	2.08+13
AC-227	Y	8.24+15	8.09+11	1.46+13	8.08+14	7.28+14	5.55+15	3.60+15	3.58+16	5.23+13	4.16+15
AC-227	W	6.72+14	5.08+11	1.98+12	1.30+15	1.97+15	1.53+16	9.76+15	9.76+16	1.12+14	5.45+15
TH-228	Y	5.73+15	2.45+12	2.49+13	1.54+14	1.23+13	2.17+13	2.98+14	3.21+15	9.08+12	1.07+15
TH-228	W	9.38+14	1.89+12	7.37+12	1.78+14	2.11+13	9.77+13	1.70+15	1.85+16	2.22+13	9.06+14
TH-229	Y	9.84+15	2.31+12	1.23+13	8.57+14	3.88+13	1.21+14	5.27+15	6.26+16	3.51+13	5.08+15
TH-229	W	9.29+14	1.77+12	4.21+12	1.14+15	3.64+13	1.77+14	1.27+16	1.52+17	3.59+13	6.25+15
RN-222	*	9.32+11	8.39+11	8.33+11	9.69+11	9.01+11	8.57+11	9.25+11	1.02+12	1.02+12	9.34+11
RA-226	W	4.48+14	5.38+10	1.46+12	3.76+13	5.31+12	5.30+12	2.00+13	1.84+14	5.31+12	6.81+13
RA-228	W	3.89+13	5.27+11	1.00+12	2.05+13	4.80+12	6.45+12	1.41+13	1.13+14	4.93+12	1.49+13
TH-230	Y	4.21+15	2.15+10	8.00+11	3.04+14	8.80+12	4.40+13	2.03+15	3.18+16	8.80+12	2.07+15
TH-230	W	4.32+14	2.21+10	7.48+11	4.48+14	2.08+13	1.04+14	4.98+15	7.82+16	2.24+13	3.02+15
TH-232	Y	3.63+15	6.12+10	8.80+11	3.04+14	8.80+12	3.92+13	2.17+15	3.62+16	8.80+12	2.13+15
TH-232	W	3.76+14	7.01+10	7.39+11	4.72+14	1.84+13	8.80+13	5.22+15	8.80+16	1.92+13	3.33+15

[#] Notation: 1.00+09 means 1.00 x 10^9

TABLE D-19 . PDCF-1: PDCF for Chronic Airborne Pathways (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	4.75+15	2.15+11	1.37+12	1.10+15	1.54+15	1.19+16	6.46+15	7.49+16	9.60+13	5.27+15
PA-231 W	4.64+14	3.28+11	1.05+12	2.09+15	3.53+15	2.74+16	1.47+16	1.73+17	2.08+14	9.28+15
U-232 Y	7.10+15	4.47+11	8.80+12	4.24+14	4.64+13	3.60+13	7.04+14	7.63+15	1.44+13	2.59+15
U-232 W	5.36+14	6.84+10	1.04+12	6.08+13	7.76+13	2.96+12	4.56+14	5.03+15	1.52+12	2.80+14
U-232 D	8.80+12	1.49+11	4.10+11	1.60+14	2.56+14	2.32+12	1.38+15	1.53+16	2.40+12	6.42+14
U-233 Y	4.34+15	3.10+10	8.80+11	1.36+14	1.36+13	1.24+11	1.92+12	2.88+13	1.31+11	8.96+14
U-233 W	4.48+14	4.97+10	7.54+11	2.08+13	4.00+13	3.55+11	5.60+12	8.00+13	3.75+11	6.24+13
U-233 D	7.60+12	1.14+11	3.30+11	4.56+13	1.36+14	1.20+12	1.84+13	2.80+14	1.28+12	2.03+13
U-234 Y	4.29+15	3.07+10	8.80+11	1.28+14	1.36+13	1.30+11	1.92+12	2.80+13	1.29+11	8.88+14
U-234 W	4.40+14	4.93+10	7.53+11	2.00+13	3.92+13	3.71+11	5.44+12	8.00+13	3.71+11	6.17+13
U-234 D	7.52+12	1.12+11	3.28+11	4.48+13	1.28+14	1.20+12	1.84+13	2.72+14	1.20+12	2.00+13
U-235 Y	3.87+15	2.80+11	2.08+12	1.20+14	1.21+13	3.36+11	1.61+12	2.33+13	2.89+11	7.98+14
U-235 W	4.00+14	1.45+11	9.60+11	1.85+13	3.53+13	4.13+11	4.33+12	6.81+13	4.66+11	5.55+13
U-235 D	6.89+12	1.96+11	4.29+11	4.09+13	1.20+14	1.12+12	1.45+13	2.24+14	1.24+12	1.73+13
U-236 Y	4.05+15	3.07+10	8.00+11	1.28+14	1.28+13	1.24+11	1.60+12	2.56+13	1.22+11	8.32+14
U-236 W	4.16+14	5.18+10	7.14+11	1.92+13	3.68+13	3.54+11	4.72+12	7.36+13	3.50+11	5.81+13
U-236 D	7.12+12	1.24+11	3.26+11	4.24+13	1.20+14	1.20+12	1.60+13	2.48+14	1.12+12	1.84+13
U-238 Y	3.84+15	1.04+12	3.19+12	1.21+14	1.30+13	1.13+12	2.65+12	2.44+13	1.46+12	7.90+14
U-238 W	3.93+14	1.02+12	1.83+12	1.95+13	3.62+13	1.31+12	5.53+12	6.76+13	1.66+12	5.56+13
U-238 D	7.69+12	1.07+12	1.26+12	4.11+13	1.21+14	2.02+12	1.63+13	2.25+14	2.45+12	1.81+13
NP-237 Y	4.55+15	4.98+11	1.56+12	5.28+14	8.16+14	6.31+15	2.40+15	3.39+16	4.66+13	2.77+15
NP-237 W	4.72+14	7.53+11	1.40+12	9.76+14	2.01+15	1.55+16	5.94+15	8.40+16	1.12+14	4.58+15
PU-236 Y	3.56+15	2.77+10	1.04+12	9.60+13	5.28+13	4.00+14	1.44+14	1.67+15	2.72+12	6.36+14
PU-236 W	5.12+14	2.33+10	8.80+11	1.20+14	2.56+14	1.98+15	6.48+14	7.54+15	1.20+13	5.27+14
PU-238 Y	4.86+15	2.32+10	9.60+11	4.80+14	7.20+14	5.60+15	2.09+15	2.62+16	4.08+13	2.46+15
PU-238 W	5.12+14	2.22+10	8.80+11	8.88+14	1.84+15	1.43+16	5.37+15	6.72+16	1.04+14	3.91+15
PU-239 Y	4.64+15	2.18+10	8.80+11	5.36+14	8.24+14	6.38+15	2.42+15	3.33+16	4.64+13	2.78+15
PU-239 W	4.80+14	2.08+10	8.00+11	9.92+14	2.04+15	1.58+16	6.05+15	8.32+16	1.20+14	4.60+15
PU-240 Y	4.63+15	2.20+10	8.80+11	5.36+14	8.24+14	6.36+15	2.42+15	3.32+16	4.64+13	2.77+15
PU-240 W	4.80+14	2.10+10	8.00+11	9.92+14	2.04+15	1.58+16	6.04+15	8.32+16	1.20+14	4.80+15
PU-241 Y	8.80+12	8.24+08	5.50+09	9.60+12	1.76+13	1.36+14	5.44+13	6.80+14	1.28+12	4.30+13
PU-241 W	1.23+11	1.45+09	4.98+09	2.00+13	3.92+13	3.04+14	1.28+14	1.55+15	2.40+12	8.72+13
PU-242 Y	4.40+15	2.20+10	8.80+11	5.04+14	7.84+14	6.06+15	2.30+15	3.35+16	4.48+13	2.69+15
PU-242 W	4.56+14	2.17+10	7.79+11	9.44+14	1.94+15	1.50+16	5.74+15	8.40+16	1.12+14	4.51+15
PU-244 Y	4.38+15	1.28+12	5.36+12	5.04+14	7.76+14	5.99+15	2.17+15	3.39+16	4.42+13	2.67+15
PU-244 W	4.48+14	1.44+12	2.56+12	9.36+14	1.92+15	1.49+16	5.41+15	8.48+16	1.12+14	4.49+15
AM-241 Y	4.92+15	8.03+10	1.05+12	5.52+14	8.48+14	6.59+15	2.50+15	3.12+16	4.80+13	2.78+15
AM-241 W	5.12+14	1.04+11	8.89+11	1.02+15	2.10+15	1.63+16	6.22+15	7.78+16	1.20+14	4.50+15
AM-243 Y	4.76+15	5.85+11	3.06+12	5.52+14	8.56+14	6.58+15	2.50+15	3.34+16	4.90+13	2.82+15
AM-243 W	4.88+14	7.87+11	1.62+12	1.02+15	2.10+15	1.62+16	6.22+15	8.24+16	1.20+14	4.63+15
CM-242 Y	1.38+15	2.16+10	9.60+11	2.40+13	6.00+12	4.64+13	1.60+13	1.84+14	3.96+11	1.86+14
CM-242 W	4.40+14	2.30+10	8.80+11	2.80+13	4.96+13	3.84+14	1.20+14	1.34+15	2.40+12	1.41+14
CM-243 Y	5.02+15	2.38+11	1.26+12	3.84+14	5.44+14	4.21+15	1.52+15	1.72+16	2.97+13	1.97+15
CM-243 W	5.44+14	3.16+11	1.18+12	6.96+14	1.46+15	1.13+16	4.10+15	4.62+16	8.01+13	2.88+15
CM-244 Y	4.86+15	2.42+10	1.04+12	3.12+14	4.16+14	3.18+15	1.13+15	1.30+16	2.16+13	1.64+15
CM-244 W	5.36+14	2.35+10	8.80+11	5.44+14	1.16+15	8.96+15	3.19+15	3.65+16	6.16+13	2.28+15
CM-248 Y	1.78+16	2.72+13	1.44+13	2.08+15	3.21+15	2.47+16	2.38+15	3.42+16	1.92+14	7.02+15
CM-248 W	1.86+15	4.00+13	2.56+13	3.86+15	7.89+15	6.08+16	5.89+15	8.48+16	4.64+14	8.56+15
CF-252 Y	6.97+15	2.56+12	2.72+12	1.68+14	9.60+13	7.52+14	1.20+14	1.30+15	5.44+12	1.10+15
CF-252 W	1.04+15	2.08+12	3.20+12	2.32+14	4.88+14	3.78+15	6.16+14	6.58+15	2.40+13	7.07+14

TABLE D-20 . PDCF-2: PDCF for Acute Airborne Pathways

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	1.00+09[#]	1.00+09	1.06+09	1.00+09	1.03+09	9.92+08	9.92+08	7.88+08	9.92+08	1.01+09
C-14	*	4.94+07	5.88+07	5.78+07	1.13+08	6.34+07	7.10+07	1.94+08	4.06+08	5.18+07	8.72+07
NA-22	D	3.55+11	3.07+11	3.14+11	3.71+11	3.54+11	3.37+11	3.88+11	4.39+11	4.13+11	3.78+11
CL-36	D	3.94+10	1.63+10	1.49+10	1.93+10	1.49+10	1.49+10	1.49+10	1.49+10	1.49+10	1.79+10
CL-36	W	1.35+12	1.97+10	1.49+10	2.50+11	1.49+10	1.49+10	1.49+10	1.49+10	1.49+10	1.75+11
FE-55	W	2.56+10	8.37+07	2.91+09	5.06+09	8.72+09	1.66+10	8.64+09	5.92+09	8.56+09	1.26+10
FE-55	Y	2.01+11	1.03+08	3.13+09	4.90+09	4.04+09	7.66+09	4.02+09	2.75+09	3.96+09	3.17+10
CO-60	W	1.35+12	4.27+11	5.03+11	4.61+11	4.54+11	5.21+11	4.49+11	4.45+11	4.97+11	6.01+11
CO-60	Y	1.07+13	1.08+12	5.08+11	9.80+11	7.73+11	1.25+12	8.25+11	7.34+11	8.63+11	2.24+12
NI-59	D	1.06+10	1.02+10	1.14+10	1.06+10	1.02+10	1.02+10	1.05+10	1.04+10	1.11+10	1.06+10
NI-59	W	3.55+10	3.14+09	6.63+09	7.40+09	3.08+09	3.08+09	3.14+09	3.12+09	3.38+09	7.37+09
NI-63	D	2.58+10	2.44+10	2.82+10	2.48+10	2.44+10	2.43+10	2.43+10	2.43+10	2.43+10	2.48+10
NI-63	W	9.12+10	7.61+09	1.99+10	1.84+10	7.34+09	7.31+09	7.31+09	7.31+09	7.31+09	1.84+10
SR-90	D	7.91+10	1.58+09	1.13+11	1.92+12	1.17+11	1.17+11	8.80+12	1.76+13	1.17+11	1.68+12
SR-90	Y	6.80+13	2.29+11	7.12+12	1.20+12	2.92+10	1.52+11	9.60+11	1.84+12	2.92+10	1.16+13
NB-94	W	1.45+12	3.01+11	4.37+11	5.14+11	4.13+11	3.09+11	4.03+11	5.04+11	3.44+11	5.06+11
NB-94	Y	2.24+13	1.14+12	4.71+11	3.54+12	7.76+11	1.36+12	8.83+11	8.17+11	9.22+11	3.53+12
TC-99	D	7.70+09	3.92+09	4.07+09	1.35+09	2.78+09	3.82+09	1.95+09	2.49+09	8.56+10	5.66+09
TC-99	W	4.18+11	4.56+09	1.33+10	7.10+09	2.46+09	3.37+09	1.72+09	2.20+09	7.57+10	5.80+10
RU-106	Y	3.04+13	8.13+10	1.15+12	5.24+11	9.87+10	1.18+11	1.03+11	1.12+11	1.08+11	4.20+12
AG-108	D	3.98+11	3.45+11	2.83+11	4.77+11	4.59+11	2.13+12	3.16+11	3.22+11	3.14+11	4.68+11
AG-108	W	1.03+12	3.03+11	3.29+11	4.39+11	3.19+11	8.52+11	2.91+11	3.03+11	3.21+11	4.30+11
AG-108	Y	1.37+13	1.12+12	3.75+11	2.50+12	7.45+11	1.66+12	8.58+11	7.51+11	8.73+11	2.49+12
CD-109	D	1.00+11	8.64+10	1.00+11	1.29+12	1.17+13	2.11+12	1.03+11	9.47+10	8.08+10	9.14+11
CD-109	W	4.33+11	2.75+10	8.80+10	4.19+11	3.38+12	6.08+11	3.07+10	2.94+10	2.47+10	3.19+11
CD-109	Y	2.31+12	1.31+10	7.48+10	5.14+11	9.93+11	1.86+11	1.38+10	1.36+10	9.14+09	3.63+11
SN-126	D	7.60+11	6.22+11	8.89+11	9.95+11	7.02+11	6.42+11	1.95+12	3.82+12	7.44+11	9.84+11
SN-126	W	4.76+12	4.39+11	1.03+12	1.10+12	4.27+11	4.45+11	7.85+11	1.31+12	5.00+11	1.09+12
SB-125	D	7.66+10	6.22+10	8.59+10	8.01+10	6.59+10	8.51+10	7.79+10	1.49+11	7.98+10	7.73+10
SB-125	W	7.00+11	7.17+10	1.51+11	1.61+11	6.69+10	7.91+10	7.46+10	9.75+10	8.27+10	1.58+11
I-129	D	2.58+10	1.46+10	1.26+10	5.32+10	3.59+10	1.91+10	1.22+10	3.86+10	4.00+13	1.23+12
CS-134	D	4.80+11	4.54+11	4.85+11	5.87+11	7.47+11	7.55+11	7.06+11	7.07+11	6.78+11	7.40+11
CS-135	D	5.12+09	3.06+08	6.81+08	3.52+10	5.98+10	5.98+10	5.98+10	6.90+10	5.98+10	5.35+10
CS-137	D	2.06+11	1.82+11	1.97+11	1.08+11	4.85+11	5.04+11	4.71+11	5.12+11	4.54+11	4.66+11
EU-152	W	1.85+12	7.24+11	5.77+11	1.93+12	1.24+12	1.04+13	2.49+12	7.27+12	4.31+11	1.92+12
EU-154	W	2.50+12	6.81+11	6.76+11	2.46+12	1.16+12	1.28+13	3.30+12	1.57+13	4.14+11	2.45+12
PB-210	W	4.96+13	9.77+09	3.76+11	2.80+13	2.64+13	2.48+13	1.76+13	1.60+14	5.36+12	2.08+13
AC-227	Y	8.24+15	6.40+11	1.45+13	8.08+14	7.28+14	5.55+15	3.60+15	3.58+16	5.21+13	4.16+15
AC-227	W	6.72+14	3.39+11	1.81+12	1.30+15	1.97+15	1.53+16	9.76+15	9.76+16	1.12+14	5.45+15
TH-228	Y	5.73+15	1.17+12	2.36+13	1.52+14	1.09+13	2.04+13	2.96+14	3.21+15	7.43+12	1.07+15
TH-228	W	9.36+14	6.04+11	6.10+12	1.76+14	1.97+13	9.64+13	1.70+15	1.85+16	2.05+13	9.04+14
TH-229	Y	9.84+15	1.48+12	1.15+13	8.56+14	3.79+13	1.20+14	5.27+15	6.26+16	3.40+13	5.08+15
TH-229	W	9.28+14	9.43+11	3.39+12	1.14+15	3.55+13	1.76+14	1.27+16	1.52+17	3.48+13	6.25+15
RN-222	*	2.56+11	2.18+11	2.16+11	2.52+11	2.34+11	2.23+11	2.40+11	2.64+11	2.67+11	2.44+11
RA-226	W	4.48+14	5.12+10	1.46+12	3.76+13	5.30+12	5.30+12	2.00+13	1.84+14	5.31+12	6.81+13
RA-228	W	3.85+13	1.89+11	6.70+11	2.01+13	4.44+12	6.11+12	1.37+13	1.12+14	4.47+12	1.45+13
TH-230	Y	4.21+15	2.14+10	8.00+11	3.04+14	8.80+12	4.40+13	2.03+15	3.18+16	8.80+12	2.07+15
TH-230	W	4.32+14	2.20+10	7.48+11	4.48+14	2.08+13	1.04+14	4.98+15	7.82+16	2.24+13	3.02+15
TH-232	Y	3.63+15	6.11+10	8.80+11	3.04+14	8.80+12	3.92+13	2.17+15	3.62+16	8.80+12	2.13+15
TH-232	W	3.76+14	7.00+10	7.38+11	4.72+14	1.84+13	8.80+13	5.22+15	8.80+16	1.92+13	3.33+15

[#] Notation: 1.00+09 means 1.00 x 10^9

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TABLE D-20 . PDCF-2: PDCF for Acute Airborne Pathways (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	4.75+15	2.04+11	1.36+12	1.10+15	1.54+15	1.19+16	6.46+15	7.49+16	9.60+13	5.27+15
PA-231 W	4.64+14	3.17+11	1.04+12	2.09+15	3.53+15	2.74+16	1.47+16	1.73+17	2.08+14	9.28+15
U-232 Y	7.10+15	4.46+11	8.80+12	4.24+14	4.64+13	3.60+13	7.04+14	7.63+15	1.44+13	2.59+15
U-232 W	5.36+14	6.84+10	1.04+12	6.08+13	7.76+13	2.96+12	4.56+14	5.03+15	1.52+12	2.80+14
U-232 D	8.80+12	1.49+11	4.10+11	1.60+14	2.56+14	2.32+12	1.38+15	1.53+16	2.40+12	6.42+14
U-233 Y	4.34+15	3.09+10	8.80+11	1.36+14	1.36+13	1.24+11	1.92+12	2.88+13	1.31+11	8.96+14
U-233 W	4.48+14	4.96+10	7.54+11	2.08+13	4.00+13	3.55+11	5.60+12	8.00+13	3.75+11	6.24+13
U-233 D	7.60+12	1.14+11	3.30+11	4.56+13	1.36+14	1.20+12	1.84+13	2.80+14	1.28+12	2.03+13
U-234 Y	4.29+15	3.07+10	8.80+11	1.28+14	1.36+13	1.30+11	1.92+12	2.80+13	1.29+11	8.88+14
U-234 W	4.40+14	4.92+10	7.53+11	2.00+13	3.92+13	3.71+11	5.44+12	8.00+13	3.70+11	6.17+13
U-234 D	7.52+12	1.12+11	3.28+11	4.48+13	1.28+14	1.20+12	1.84+13	2.72+14	1.20+12	2.00+13
U-235 Y	3.87+15	2.19+11	2.02+12	1.20+14	1.20+13	2.73+11	1.54+12	2.32+13	1.96+11	7.98+14
U-235 W	4.00+14	8.30+10	9.00+11	1.84+13	3.52+13	3.50+11	4.26+12	6.80+13	3.73+11	5.55+13
U-235 D	6.82+12	1.34+11	3.69+11	4.08+13	1.20+14	1.06+12	1.44+13	2.24+14	1.15+12	1.72+13
U-236 Y	4.05+15	3.07+10	8.00+11	1.28+14	1.28+13	1.24+11	1.60+12	2.56+13	1.22+11	8.32+14
U-236 W	4.16+14	5.18+10	7.14+11	1.92+13	3.68+13	3.54+11	4.72+12	7.36+13	3.50+11	5.81+13
U-236 D	7.12+12	1.24+11	3.26+11	4.24+13	1.20+14	1.20+12	1.60+13	2.48+14	1.12+12	1.84+13
U-238 Y	3.84+15	3.13+11	2.48+12	1.20+14	1.23+13	3.85+11	1.86+12	2.35+13	4.58+11	7.89+14
U-238 W	3.92+14	2.91+11	1.12+12	1.87+13	3.55+13	5.66+11	4.74+12	6.67+13	6.63+11	5.47+13
U-238 D	6.90+12	3.42+11	5.42+11	4.03+13	1.20+14	1.28+12	1.55+13	2.24+14	1.45+12	1.73+13
NP-237 Y	4.55+15	4.09+11	1.47+12	5.28+14	8.16+14	6.31+15	2.40+15	3.39+16	4.64+13	2.77+15
NP-237 W	4.72+14	6.64+11	1.31+12	9.76+14	2.01+15	1.55+16	5.94+15	8.40+16	1.12+14	4.58+15
PU-236 Y	3.56+15	2.76+10	1.04+12	9.60+13	5.28+13	4.00+14	1.44+14	1.67+15	2.72+12	6.36+14
PU-236 W	5.12+14	2.32+10	8.80+11	1.20+14	2.56+14	1.98+15	6.48+14	7.54+15	1.20+13	5.27+14
PU-238 Y	4.86+15	2.32+10	9.60+11	4.80+14	7.20+14	5.60+15	2.09+15	2.62+16	4.08+13	2.46+15
PU-238 W	5.12+14	2.22+10	8.80+11	8.88+14	1.84+15	1.43+16	5.37+15	6.72+16	1.04+14	3.91+15
PU-239 Y	4.64+15	2.18+10	8.80+11	5.36+14	8.24+14	6.38+15	2.42+15	3.33+16	4.64+13	2.78+15
PU-239 W	4.80+14	2.07+10	8.00+11	9.92+14	2.04+15	1.58+16	6.05+15	8.32+16	1.20+14	4.60+15
PU-240 Y	4.63+15	2.19+10	8.80+11	5.36+14	8.24+14	6.36+15	2.42+15	3.32+16	4.64+13	2.77+15
PU-240 W	4.80+14	2.10+10	8.00+11	9.92+14	2.04+15	1.58+16	6.04+15	8.32+16	1.20+14	4.80+15
PU-241 Y	8.80+12	8.24+08	5.50+09	9.60+12	1.76+13	1.36+14	5.44+13	6.80+14	1.28+12	4.30+13
PU-241 W	1.23+11	1.45+09	4.98+09	2.00+13	3.92+13	3.04+14	1.28+14	1.55+15	2.40+12	8.72+13
PU-242 Y	4.40+15	2.20+10	8.80+11	5.04+14	7.84+14	6.06+15	2.30+15	3.35+16	4.48+13	2.69+15
PU-242 W	4.56+14	2.17+10	7.79+11	9.44+14	1.94+15	1.50+16	5.74+15	8.40+16	1.12+14	4.51+15
PU-244 Y	4.38+15	1.16+12	5.24+12	5.04+14	7.76+14	5.99+15	2.17+15	3.39+16	4.41+13	2.67+15
PU-244 W	4.48+14	1.32+12	2.44+12	9.36+14	1.92+15	1.49+16	5.41+15	8.48+16	1.12+14	4.49+15
AM-241 Y	4.92+15	7.28+10	1.04+12	5.52+14	8.48+14	6.59+15	2.50+15	3.12+16	4.80+13	2.78+15
AM-241 W	5.12+14	9.68+10	8.82+11	1.02+15	2.10+15	1.63+16	6.22+15	7.78+16	1.20+14	4.50+15
AM-243 Y	4.76+15	5.04+11	2.99+12	5.52+14	8.56+14	6.58+15	2.50+15	3.34+16	4.88+13	2.82+15
AM-243 W	4.88+14	7.05+11	1.55+12	1.02+15	2.10+15	1.62+16	6.22+15	8.24+16	1.20+14	4.63+15
CM-242 Y	1.38+15	2.16+10	9.60+11	2.40+13	6.00+12	4.64+13	1.60+13	1.84+14	3.96+11	1.86+14
CM-242 W	4.40+14	2.30+10	8.80+11	2.80+13	4.96+13	3.84+14	1.20+14	1.34+15	2.40+12	1.41+14
CM-243 Y	5.02+15	1.90+11	1.22+12	3.84+14	5.44+14	4.21+15	1.52+15	1.72+16	2.96+13	1.97+15
CM-243 W	5.44+14	2.68+11	1.14+12	6.96+14	1.46+15	1.13+16	4.10+15	4.62+16	8.00+13	2.88+15
CM-244 Y	4.86+15	2.42+10	1.04+12	3.12+14	4.16+14	3.18+15	1.13+15	1.30+16	2.16+13	1.64+15
CM-244 W	5.36+14	2.34+10	8.80+11	5.44+14	1.16+15	8.96+15	3.19+15	3.65+16	6.16+13	2.28+15
CM-248 Y	1.78+16	2.72+13	1.44+13	2.08+15	3.21+15	2.47+16	2.38+15	3.42+16	1.92+14	7.02+15
CM-248 W	1.86+15	4.00+13	2.56+13	3.86+15	7.89+15	6.08+16	5.89+15	8.48+16	4.64+14	8.56+15
CF-252 Y	6.97+15	2.56+12	2.72+12	1.68+14	9.60+13	7.52+14	1.20+14	1.30+15	5.44+12	1.10+15
CF-252 W	1.04+15	2.08+12	3.20+12	2.32+14	4.88+14	3.78+15	6.16+14	6.58+15	[illegible]40+13	7.07+14

TABLE D-21 . PDCF-3: PDCF for Chronic Airborne Pathways including Food/Air Pathways

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	5.28+10#	6.79+10	8.97+10	5.24+10	5.41+10	5.23+10	5.22+10	4.14+10	5.23+10	5.67+10
C-14	*	7.32+11	1.04+12	1.26+12	1.66+12	9.14+11	1.06+12	2.92+12	6.08+12	7.67+11	1.33+12
NA-22	D	6.85+12	3.37+12	4.05+12	4.26+12	4.46+12	4.21+12	5.54+12	6.88+12	4.12+12	5.34+12
CL-36	D	3.56+12	4.90+12	3.53+12	3.66+12	3.53+12	3.53+12	3.53+12	3.53+12	3.53+12	3.62+12
CL-36	W	4.87+12	4.90+12	3.53+12	3.89+12	3.53+12	3.53+12	3.53+12	3.53+12	3.53+12	3.78+12
FE-55	W	9.07+10	1.77+09	6.78+10	3.72+10	6.77+10	1.28+11	6.74+10	4.60+10	6.65+10	8.40+10
FE-55	Y	2.66+11	1.79+09	6.80+10	3.70+10	6.31+10	1.19+11	6.28+10	4.28+10	6.19+10	1.03+11
CO-60	W	2.99+12	1.72+12	4.71+12	1.80+12	1.85+12	1.96+12	1.83+12	1.76+12	1.91+12	2.49+12
CO-60	Y	1.24+13	2.37+12	4.72+12	2.32+12	2.16+12	2.69+12	2.21+12	2.05+12	2.27+12	4.13+12
NI-59	D	2.02+10	2.13+10	8.47+10	2.62+10	1.99+10	1.99+10	2.04+10	2.03+10	2.17+10	2.61+10
NI-59	W	4.51+10	1.43+10	8.00+10	2.30+10	1.28+10	1.28+10	1.31+10	1.30+10	1.40+10	2.28+10
NI-63	D	4.90+10	5.30+10	2.78+11	6.72+10	4.75+10	4.74+10	4.74+10	4.74+10	4.74+10	6.72+10
NI-63	W	1.14+11	3.62+10	2.70+11	6.08+10	3.05+10	3.04+10	3.04+10	3.04+10	3.04+10	6.08+10
SR-90	D	5.36+11	6.84+10	6.04+12	9.13+12	5.74+11	5.52+11	4.16+13	8.32+13	5.74+11	8.35+12
SR-90	Y	6.80+13	2.96+11	1.44+13	1.58+12	5.21+10	1.74+11	2.60+12	5.12+12	5.21+10	1.25+13
NB-94	W	2.35+12	2.05+12	1.98+13	4.09+12	2.09+12	1.33+12	2.16+12	2.36+12	1.33+12	4.06+12
NB-94	Y	2.33+13	2.89+12	1.99+13	7.12+12	2.45+12	2.38+12	2.64+12	2.67+12	1.91+12	7.09+12
TC-99	D	7.10+10	1.90+11	6.43+11	4.41+10	9.43+10	1.29+11	6.63+10	8.44+10	2.90+12	2.09+11
TC-99	W	4.81+11	1.90+11	6.52+11	4.98+10	9.39+10	1.29+11	6.60+10	8.41+10	2.89+12	2.62+11
RU-106	Y	3.53+13	3.74+12	1.46+14	3.93+12	4.79+12	4.81+12	4.83+12	5.55+12	4.71+12	2.04+13
AG-108	D	1.32+12	1.39+12	4.08+12	2.05+12	1.55+12	5.63+12	1.27+12	1.24+12	1.21+12	2.01+12
AG-108	W	1.95+12	1.35+12	4.12+12	2.01+12	1.41+12	4.34+12	1.25+12	1.22+12	1.22+12	1.98+12
AG-108	Y	1.47+13	2.17+12	4.17+12	4.07+12	1.84+12	5.15+12	1.81+12	1.67+12	1.77+12	4.04+12
CD-109	D	2.12+11	2.28+11	1.68+12	2.98+12	2.56+13	4.63+12	2.31+11	2.12+11	1.81+11	2.13+12
CD-109	W	5.45+11	1.69+11	1.67+12	2.10+12	1.73+13	3.13+12	1.59+11	1.47+11	1.25+11	1.53+12
CD-109	Y	2.42+12	1.55+11	1.65+12	2.20+12	1.49+13	2.70+12	1.42+11	1.31+11	1.09+11	1.58+12
SN-126	D	1.96+12	2.38+12	2.85+13	5.15+12	2.01+12	1.83+12	4.48+12	7.92+12	2.12+12	5.11+12
SN-126	W	5.96+12	2.20+12	2.87+13	5.26+12	1.74+12	1.64+12	3.31+12	5.41+12	1.88+12	5.21+12
SB-125	D	2.68+11	3.04+11	1.82+12	4.79+11	2.69+11	3.16+11	3.18+11	5.17+11	3.14+11	4.67+11
SB-125	W	8.79+11	3.03+11	2.03+12	5.59+11	2.57+11	2.59+11	2.86+11	3.30+11	3.06+11	5.47+11
I-129	D	1.02+12	1.48+11	1.27+11	3.78+12	9.35+11	8.90+11	1.11+12	1.14+12	8.91+15	2.67+14
CS-134	D	1.64+13	5.81+12	6.55+12	7.80+12	1.09+13	1.09+13	1.02+13	9.89+12	8.94+12	1.21+13
CS-135	D	1.08+12	2.33+10	5.18+10	6.67+11	1.13+12	1.13+12	1.13+12	1.31+12	1.14+12	1.13+12
CS-137	D	1.00+13	2.48+12	2.88+12	5.05+12	8.10+12	8.29+12	7.76+12	8.41+12	7.17+12	8.54+12
EU-152	W	2.36+12	1.31+12	3.72+12	2.88+12	1.81+12	1.16+13	3.18+12	8.35+12	1.02+12	2.86+12
EU-154	W	3.05+12	1.35+12	6.05+12	3.69+12	1.76+12	1.43+13	4.10+12	1.75+13	1.04+12	3.66+12
PB-210	W	7.04+13	2.38+10	1.78+12	1.46+14	9.15+13	1.22+14	8.69+13	8.25+14	2.62+13	7.47+13
AC-227	Y	8.26+15	8.62+11	1.81+13	9.93+14	1.01+15	7.82+15	5.02+15	4.99+16	6.79+13	4.94+15
AC-227	W	6.88+14	5.61+11	5.49+12	1.48+15	2.25+15	1.75+16	1.12+16	1.12+17	1.28+14	6.23+15
TH-228	Y	5.73+15	2.79+12	5.71+13	1.57+14	1.28+13	2.33+13	3.24+14	3.49+15	9.59+12	1.09+15
TH-228	W	9.38+14	2.22+12	3.96+13	1.81+14	2.16+13	9.93+13	1.72+15	1.88+16	2.27+13	9.20+14
TH-229	Y	9.84+15	2.64+12	2.67+13	8.73+14	3.94+13	1.24+14	5.45+15	6.48+16	3.58+13	5.17+15
TH-229	W	9.30+14	2.11+12	1.86+13	1.15+15	3.70+13	1.80+14	1.29+16	1.54+17	3.66+13	6.34+15
RN-222	*	9.32+11	8.39+11	8.33+11	9.69+11	9.01+11	8.57+11	9.25+11	1.02+12	1.02+12	9.34+11
RA-226	W	4.90+14	4.32+11	2.47+13	2.77+14	4.69+13	4.69+13	1.75+14	1.59+15	4.69+13	1.65+14
RA-228	W	6.78+13	6.92+11	6.03+12	1.40+14	3.30+13	3.46+13	1.06+14	8.03+14	3.31+13	7.05+13
TH-230	Y	4.21+15	2.98+11	1.31+13	3.10+14	9.10+12	4.55+13	2.10+15	3.29+16	9.11+12	2.12+15
TH-230	W	4.32+14	2.98+11	1.31+13	4.54+14	2.11+13	1.05+14	5.05+15	7.93+16	2.27+13	3.07+15
TH-232	Y	3.63+15	2.99+11	1.12+13	3.11+14	9.06+12	4.05+13	2.24+15	3.75+16	9.07+12	2.18+15
TH-232	W	3.76+14	3.08+11	1.10+13	4.79+14	1.87+13	8.93+13	5.30+15	8.92+16	1.95+13	3.38+15

#Notation: 5.28+10 means 5.28 x 10^{10}

TABLE D-21 . PDCF-3: PDCF for Chronic Airborne Pathways including Food/Air Pathways (contin

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	4.79+15	9.97+11	3.30+13	1.45+15	2.13+15	1.66+16	8.96+15	1.04+17	1.33+14	6.86+15
PA-231 W	5.01+14	1.11+12	3.27+13	2.44+15	4.13+15	3.21+16	1.72+16	2.02+17	2.45+14	1.09+16
U-232 Y	7.10+15	7.70+11	2.27+13	4.30+14	5.54+13	3.61+13	7.53+14	8.17+15	1.45+13	2.62+15
U-232 W	5.38+14	5.14+11	1.49+13	2.07+14	3.07+14	4.95+12	1.71+15	1.86+16	3.62+12	8.49+14
U-232 D	1.11+13	5.95+11	1.43+13	3.06+14	4.85+14	4.31+12	2.63+15	2.88+16	4.50+12	1.21+15
U-233 Y	4.34+15	3.21+11	1.34+13	1.38+14	1.83+13	1.66+11	2.59+12	3.85+13	1.76+11	8.98+14
U-233 W	4.49+14	4.32+11	1.26+13	6.11+13	1.58+14	1.40+12	2.23+13	3.23+14	1.49+12	8.05+13
U-233 D	8.81+12	4.96+11	1.21+13	8.59+13	2.54+14	2.25+12	3.51+13	5.23+14	2.39+12	3.84+13
U-234 Y	4.29+15	3.20+11	1.34+13	1.30+14	1.82+13	1.74+11	2.57+12	3.77+13	1.73+11	8.90+14
U-234 W	4.41+14	4.30+11	1.26+13	6.03+13	1.57+14	1.47+12	2.14+13	3.23+14	1.47+12	7.95+13
U-234 D	8.71+12	4.93+11	1.21+13	8.51+13	2.46+14	2.30+12	3.44+13	5.15+14	2.30+12	3.78+13
U-235 Y	3.87+15	5.62+11	1.53+13	1.22+14	1.63+13	3.78+11	2.12+12	3.17+13	3.30+11	7.99+14
U-235 W	4.01+14	5.18+11	1.35+13	5.46+13	1.39+14	1.37+12	1.68+13	2.69+14	1.47+12	7.11+13
U-235 D	8.02+12	5.68+11	1.29+13	7.70+13	2.24+14	2.08+12	2.70+13	4.26+14	2.25+12	3.29+13
U-236 Y	4.05+15	3.03+11	1.26+13	1.30+14	1.72+13	1.65+11	2.16+12	3.46+13	1.63+11	8.34+14
U-236 W	4.17+14	4.10+11	1.18+13	5.67+13	1.48+14	1.39+12	1.86+13	2.89+14	1.38+12	7.45+13
U-236 D	8.25+12	4.82+11	1.14+13	7.99+13	2.31+14	2.23+12	2.99+13	4.63+14	2.15+12	3.49+13
U-238 Y	3.84+15	1.30+12	1.50+13	1.23+14	1.72+13	1.16+12	3.18+12	3.20+13	1.50+12	7.91+14
U-238 W	3.94+14	1.36+12	1.29+13	5.49+13	1.40+14	2.24+12	1.87+13	2.62+14	2.64+12	7.08+13
U-238 D	8.75+12	1.41+12	1.24+13	7.65+13	2.25+14	2.95+12	2.94+13	4.20+14	3.43+12	3.34+13
NP-237 Y	4.56+15	8.78+11	1.61+13	6.01+14	9.68+14	7.47+15	2.84+15	4.02+16	5.53+13	3.11+15
NP-237 W	4.81+14	1.13+12	1.59+13	1.05+15	2.16+15	1.67+16	6.39+15	9.03+16	1.21+14	4.92+15
PU-236 Y	3.56+15	3.67+11	1.61+13	9.63+13	5.34+13	4.04+14	1.45+14	1.69+15	2.75+12	6.38+14
PU-236 W	5.12+14	3.63+11	1.59+13	1.20+14	2.57+14	1.98+15	6.49+14	7.55+15	1.20+13	5.29+14
PU-238 Y	4.86+15	3.46+11	1.53+13	4.82+14	7.24+14	5.63+15	2.10+15	2.63+16	4.10+13	2.47+15
PU-238 W	5.12+14	3.45+11	1.53+13	8.90+14	1.84+15	1.44+16	5.38+15	6.73+16	1.04+14	3.92+15
PU-239 Y	4.64+15	3.25+11	1.46+13	5.38+14	8.28+14	6.41+15	2.44+15	3.35+16	4.67+13	2.79+15
PU-239 W	4.80+14	3.23+11	1.45+13	9.94+14	2.04+15	1.59+16	6.06+15	8.34+16	1.20+14	4.61+15
PU-240 Y	4.63+15	3.25+11	1.46+13	5.38+14	8.28+14	6.39+15	2.43+15	3.34+16	4.67+13	2.78+15
PU-240 W	4.80+14	3.24+11	1.45+13	9.94+14	2.04+15	1.58+16	6.05+15	8.34+16	1.20+14	4.81+15
PU-241 Y	8.81+12	2.35+09	7.34+10	9.64+12	1.77+13	1.37+14	5.47+13	6.83+14	1.29+12	4.32+13
PU-241 W	1.28+11	2.98+09	7.29+10	2.00+13	3.93+13	3.05+14	1.28+14	1.56+15	2.41+12	8.74+13
PU-242 Y	4.40+15	3.10+11	1.39+13	5.06+14	7.88+14	6.09+15	2.31+15	3.37+16	4.50+13	2.70+15
PU-242 W	4.56+14	3.10+11	1.38+13	9.46+14	1.94+15	1.51+16	5.76+15	8.42+16	1.12+14	4.52+15
PU-244 Y	4.38+15	1.60+12	2.59+13	5.06+14	7.80+14	6.02+15	2.18+15	3.41+16	4.45+13	2.68+15
PU-244 W	4.48+14	1.76+12	2.31+13	9.38+14	1.92+15	1.49+16	5.42+15	8.50+16	1.12+14	4.50+15
AM-241 Y	4.93+15	4.35+11	1.64+13	6.25+14	1.01+15	7.84+15	2.96+15	3.71+16	5.68+13	3.12+15
AM-241 W	5.21+14	4.59+11	1.63+13	1.10+15	2.27+15	1.76+16	6.68+15	8.37+16	1.29+14	4.84+15
AM-243 Y	4.77+15	9.77+11	1.92+13	6.25+14	1.02+15	7.83+15	2.97+15	3.96+16	5.85+13	3.17+15
AM-243 W	4.98+14	1.18+12	1.77+13	1.10+15	2.27+15	1.75+16	6.68+15	8.86+16	1.30+14	4.98+15
CM-242 Y	1.38+15	3.81+11	1.67+13	2.58+13	9.91+12	7.65+13	2.56+13	2.87+14	5.82+11	1.94+14
CM-242 W	4.40+14	3.82+11	1.66+13	2.98+13	5.35+13	4.14+14	1.30+14	1.45+15	2.59+12	1.49+14
CM-243 Y	5.02+15	6.43+11	1.84+13	4.33+14	6.47+14	5.03+15	1.81+15	2.05+16	3.53+13	2.17+15
CM-243 W	5.50+14	7.21+11	1.83+13	7.45+14	1.56+15	1.21+16	4.39+15	4.95+16	8.57+13	3.08+15
CM-244 Y	4.86+15	3.64+11	1.61+13	3.50+14	4.98+14	3.82+15	1.35+15	1.56+16	2.60+13	1.80+15
CM-244 W	5.40+14	3.63+11	1.59+13	5.82+14	1.24+15	9.60+15	3.42+15	3.91+16	6.60+13	2.44+15
CM-248 Y	1.78+16	3.25+13	8.97+13	2.35+15	3.76+15	2.90+16	2.79+15	4.03+16	2.25+14	7.62+15
CM-248 W	1.90+15	4.53+13	1.01+14	4.13+15	8.44+15	6.51+16	6.31+15	9.08+16	4.97+14	9.16+15
CF-252 Y	6.97+15	4.00+12	4.17+13	1.84+14	1.32+14	1.03+15	1.64+14	1.78+15	7.16+12	1.15+15
CF-252 W	1.04+15	3.52+12	4.22+13	2.48+14	5.24+14	4.06+15	6.61+14	7.05+15	2.57+13	7.53+14

TABLE D-22 . PDCF-4: PDCF for Food/Soil Pathways

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	7.18+04[#]	9.27+04	1.23+05	7.13+04	7.35+04	7.11+04	7.09+04	5.63+04	7.11+04	7.71+04
C-14	*	1.04+06	1.48+06	1.78+06	2.34+06	1.29+06	1.50+06	4.13+06	8.61+06	1.09+06	1.88+06
NA-22	D	4.52+04	1.83+04	2.37+04	2.39+04	2.62+04	2.46+04	3.43+04	4.40+04	2.13+04	3.27+04
CL-36	D	4.91+06	6.82+06	4.91+06	5.07+06	4.91+06	4.91+06	4.91+06	4.91+06	4.91+06	5.02+06
CL-36	W	4.91+06	6.82+06	4.91+06	5.07+06	4.91+06	4.91+06	4.91+06	4.91+06	4.91+06	5.02+06
FE-55	W	4.77+01	1.24+00	4.76+01	2.35+01	4.33+01	8.20+01	4.31+01	2.94+01	4.25+01	5.24+01
FE-55	Y	4.77+01	1.24+00	4.76+01	2.35+01	4.33+01	8.20+01	4.31+01	2.94+01	4.25+01	5.24+01
CO-60	W	1.86+04	1.14+04	8.68+04	9.43+03	1.22+04	1.47+04	1.17+04	8.61+03	6.69+03	2.44+04
CO-60	Y	1.86+04	1.14+04	8.68+04	9.43+03	1.22+04	1.47+04	1.17+04	8.61+03	6.69+03	2.44+04
NI-59	D	3.43+02	4.01+02	2.63+03	5.53+02	3.48+02	3.48+02	3.56+02	3.53+02	3.79+02	5.53+02
NI-59	W	3.43+02	4.01+02	2.63+03	5.53+02	3.48+02	3.48+02	3.56+02	3.53+02	3.79+02	5.53+02
NI-63	D	8.30+02	1.03+03	8.96+03	1.52+03	8.30+02	8.30+02	8.30+02	8.30+02	8.30+02	1.52+03
NI-63	W	8.30+02	1.03+03	8.96+03	1.52+03	8.30+02	8.30+02	8.30+02	8.30+02	8.30+02	1.52+03
SR-90	D	5.55+04	8.11+03	7.20+05	8.75+05	5.55+04	5.29+04	3.98+06	7.96+06	5.55+04	8.10+05
SR-90	Y	2.78+03	8.11+03	8.89+05	4.65+04	2.78+03	2.65+03	1.99+05	3.98+05	2.78+03	1.05+05
NB-94	W	4.20+03	1.88+04	3.06+05	4.71+04	1.70+04	6.87+03	1.80+04	1.87+04	3.00+03	4.71+04
NB-94	Y	4.20+03	1.88+04	3.06+05	4.71+04	1.70+04	6.87+03	1.80+04	1.87+04	3.00+03	4.71+04
TC-99	D	5.64+04	1.65+05	5.69+05	3.81+04	8.15+04	1.12+05	5.73+04	7.29+04	2.51+06	1.81+05
TC-99	W	5.64+04	1.65+05	5.69+05	3.81+04	8.15+04	1.12+05	5.73+04	7.29+04	2.51+06	1.81+05
RU-106	Y	1.16+04	8.74+03	3.54+05	8.10+03	1.12+04	1.13+04	1.13+04	1.30+04	1.10+04	3.93+04
AG-108	D	5.67+04	9.80+04	7.23+05	1.94+05	9.97+04	6.49+05	6.23+04	3.33+04	1.22+04	1.94+05
AG-108	W	5.67+04	9.80+04	7.23+05	1.94+05	9.97+04	6.49+05	6.23+04	3.33+04	1.22+04	1.94+05
AG-108	Y	5.67+04	9.80+04	7.23+05	1.94+05	9.97+04	6.49+05	6.23+04	3.33+04	1.22+04	1.94+05
CD-109	D	4.29+04	5.54+04	6.28+05	6.68+05	5.54+06	1.00+06	5.03+04	4.44+04	3.74+04	4.81+05
CD-109	W	4.29+04	5.54+04	6.28+05	6.68+05	5.54+06	1.00+06	5.03+04	4.44+04	3.74+04	4.81+05
CD-109	Y	4.29+04	5.54+04	6.28+05	6.68+05	5.54+06	1.00+06	5.03+04	4.44+04	3.74+04	4.81+05
SN-126	D	1.62+03	4.33+03	1.17+05	1.43+04	2.22+03	1.85+03	7.39+03	1.37+04	1.49+03	1.43+04
SN-126	W	1.62+03	4.33+03	1.17+05	1.43+04	2.22+03	1.85+03	7.39+03	1.37+04	1.49+03	1.43+04
SB-125	D	3.13+02	1.54+03	3.00+04	3.94+03	5.87+02	1.29+03	1.17+03	3.05+03	2.40+02	3.94+03
SB-125	W	7.06+01	1.33+03	3.27+04	3.93+03	3.41+02	3.26+02	6.29+02	4.70+02	2.89+01	3.93+03
I-129	D	5.84+02	5.58+01	4.77+01	2.26+03	4.99+02	5.15+02	6.70+02	6.25+02	5.55+06	1.66+05
CS-134	D	1.33+05	4.15+04	4.78+04	5.68+04	8.31+04	8.31+04	7.69+04	7.36+04	6.49+04	9.31+04
CS-135	D	9.31+03	1.99+02	4.44+02	5.49+03	9.31+03	9.31+03	9.31+03	1.08+04	9.39+03	9.39+03
CS-137	D	8.31+04	1.81+04	2.15+04	4.08+04	6.42+04	6.54+04	6.13+04	6.64+04	5.58+04	6.80+04
EU-152	W	2.87+02	7.68+02	1.19+04	2.09+03	5.55+02	3.58+03	1.10+03	2.49+03	7.94+01	2.09+03
EU-154	W	2.58+02	9.78+02	2.15+04	3.08+03	5.36+02	4.42+03	1.37+03	5.32+03	6.81+01	3.08+03
PB-210	W	1.45+05	9.40+01	9.79+03	8.20+05	4.53+05	6.75+05	4.82+05	4.63+06	1.45+05	3.75+05
AC-227	Y	6.79+04	2.30+02	1.53+04	8.03+05	1.24+06	9.88+06	6.18+06	6.18+07	6.79+04	3.41+06
AC-227	W	6.79+04	2.30+02	1.53+04	8.03+05	1.24+06	9.88+06	6.18+06	6.18+07	6.79+04	3.41+06
TH-228	Y	3.75+03	2.46+03	2.35+05	1.90+04	3.94+03	1.17+04	1.90+05	2.05+06	3.71+03	1.05+05
TH-228	W	3.75+03	2.46+03	2.35+05	1.90+04	3.94+03	1.17+04	1.90+05	2.05+06	3.71+03	1.05+05
TH-229	Y	5.06+03	2.44+03	1.05+05	1.15+05	4.73+03	1.82+04	1.30+06	1.60+07	4.70+03	6.46+05
TH-229	W	5.06+03	2.44+03	1.05+05	1.15+05	4.73+03	1.82+04	1.30+06	1.60+07	4.70+03	6.46+05
RN-222	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
RA-226	W	1.00+06	9.12+03	5.60+05	5.77+06	1.00+06	1.00+06	3.74+06	3.40+07	1.00+06	2.34+06
RA-228	W	6.96+05	3.97+03	1.21+05	2.89+06	6.79+05	6.79+05	2.21+06	1.66+07	6.79+05	1.34+06
TH-230	Y	2.28+03	2.02+03	9.01+04	4.63+04	2.16+03	1.09+04	5.01+05	8.01+06	2.28+03	3.15+05
TH-230	W	2.28+03	2.02+03	9.01+04	4.63+04	2.16+03	1.09+04	5.01+05	8.01+06	2.28+03	3.15+05
TH-232	Y	1.99+03	1.74+03	7.51+04	4.82+04	1.88+03	9.41+03	5.51+05	9.01+06	1.97+03	3.46+05
TH-232	W	1.99+03	1.74+03	7.51+04	4.82+04	1.88+03	9.41+03	5.51+05	9.01+06	1.97+03	3.46+05

[#] Notation: 7.18+04 means 7.18 x 10^4

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TABLE D-22 . PDCF-4: PDCF for Food/Soil Pathways (continued)

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231	Y	1.59+05	3.40+03	1.38+05	1.52+06	2.61+06	2.03+07	1.09+07	1.29+08	1.59+05	6.92+06
PA-231	W	1.59+05	3.40+03	1.38+05	1.52+06	2.61+06	2.03+07	1.09+07	1.29+08	1.59+05	6.92+06
U-232	Y	3.93+02	1.41+03	6.05+04	2.54+04	3.93+04	3.48+02	2.15+05	2.36+06	3.66+02	1.04+05
U-232	W	9.80+03	1.94+03	6.05+04	6.35+05	9.98+05	8.65+03	5.44+06	5.90+07	9.13+03	2.48+06
U-232	D	9.80+03	1.94+03	6.05+04	6.35+05	9.98+05	8.65+03	5.44+06	5.90+07	9.13+03	2.48+06
U-233	Y	2.10+02	1.26+03	5.44+04	7.26+03	2.05+04	1.83+02	2.90+03	4.23+04	1.93+02	7.65+03
U-233	W	5.26+03	1.67+03	5.14+04	1.75+05	5.14+05	4.57+03	7.26+04	1.06+06	4.84+03	7.89+04
U-233	D	5.26+03	1.67+03	5.14+04	1.75+05	5.14+05	4.57+03	7.26+04	1.06+06	4.84+03	7.89+04
U-234	Y	2.08+02	1.26+03	5.44+04	7.20+03	2.02+04	1.91+02	2.82+03	4.23+04	1.91+02	7.59+03
U-234	W	5.20+03	1.66+03	5.14+04	1.75+05	5.14+05	4.78+03	6.96+04	1.06+06	4.78+03	7.77+04
U-234	D	5.20+03	1.66+03	5.14+04	1.75+05	5.14+05	4.78+03	6.96+04	1.06+06	4.78+03	7.77+04
U-235	Y	2.25+02	1.22+03	5.75+04	6.53+03	1.83+04	1.84+02	2.23+03	3.63+04	1.75+02	7.38+03
U-235	W	4.90+03	1.62+03	5.44+04	1.57+05	4.54+05	4.17+03	5.44+04	8.77+05	4.39+03	6.77+04
U-235	D	4.90+03	1.62+03	5.44+04	1.57+05	4.54+05	4.17+03	5.44+04	8.77+05	4.39+03	6.77+04
U-236	Y	1.96+02	1.19+03	5.14+04	6.77+03	1.91+04	1.80+02	2.43+03	3.93+04	1.80+02	7.11+03
U-236	W	4.90+03	1.56+03	4.84+04	1.63+05	4.84+05	4.51+03	6.05+04	9.38+05	4.51+03	7.17+04
U-236	D	4.90+03	1.56+03	4.84+04	1.63+05	4.84+05	4.51+03	6.05+04	9.38+05	4.51+03	7.17+04
U-238	Y	1.86+02	1.11+03	5.14+04	6.41+03	1.80+04	1.62+02	2.31+03	3.33+04	1.70+02	6.77+03
U-238	W	4.63+03	1.47+03	4.84+04	1.54+05	4.54+05	4.05+03	5.75+04	8.47+05	4.26+03	6.65+04
U-238	D	4.63+03	1.47+03	4.84+04	1.54+05	4.54+05	4.05+03	5.75+04	8.47+05	4.26+03	6.65+04
NP-237	Y	8.58+05	3.75+04	1.43+06	7.15+06	1.50+07	1.14+08	4.36+07	6.22+08	8.58+05	3.35+07
NP-237	W	8.58+05	3.75+04	1.43+06	7.15+06	1.50+07	1.14+08	4.36+07	6.22+08	8.58+05	3.35+07
PU-236	Y	2.57+01	3.32+02	1.47+04	3.00+02	5.44+02	4.21+03	1.39+03	1.61+04	2.57+01	2.22+03
PU-236	W	2.57+01	3.32+02	1.47+04	3.00+02	5.44+02	4.21+03	1.39+03	1.61+04	2.57+01	2.22+03
PU-238	Y	2.16+02	3.15+02	1.41+04	1.90+03	3.83+03	2.95+04	1.14+04	1.41+05	2.16+02	9.17+03
PU-238	W	2.16+02	3.15+02	1.41+04	1.90+03	3.83+03	2.95+04	1.14+04	1.41+05	2.16+02	9.17+03
PU-239	Y	2.43+02	2.96+02	1.34+04	2.10+03	4.24+03	3.28+04	1.27+04	1.74+05	2.43+02	1.05+04
PU-239	W	2.43+02	2.96+02	1.34+04	2.10+03	4.24+03	3.28+04	1.27+04	1.74+05	2.43+02	1.05+04
PU-240	Y	2.42+02	2.96+02	1.34+04	2.10+03	4.23+03	3.28+04	1.27+04	1.74+05	2.42+02	1.05+04
PU-240	W	2.42+02	2.96+02	1.34+04	2.10+03	4.23+03	3.28+04	1.27+04	1.74+05	2.42+02	1.05+04
PU-241	Y	5.03+00	1.49+00	6.64+01	4.15+01	8.24+01	6.36+02	2.58+02	3.23+03	5.02+00	1.87+02
PU-241	W	5.03+00	1.49+00	6.64+01	4.15+01	8.24+01	6.36+02	2.58+02	3.23+03	5.02+00	1.87+02
PU-242	Y	2.31+02	2.82+02	1.27+04	2.00+03	4.02+03	3.15+04	1.21+04	1.74+05	2.31+02	1.02+04
PU-242	W	2.31+02	2.82+02	1.27+04	2.00+03	4.02+03	3.15+04	1.21+04	1.74+05	2.31+02	1.02+04
PU-244	Y	2.48+02	3.13+02	2.01+04	2.02+03	4.02+03	3.08+04	1.14+04	1.74+05	2.30+02	1.08+04
PU-244	W	2.48+02	3.13+02	2.01+04	2.02+03	4.02+03	3.08+04	1.14+04	1.74+05	2.30+02	1.08+04
AM-241	Y	8.54+04	3.45+03	1.49+05	7.12+05	1.57+06	1.21+07	4.55+06	5.69+07	8.54+04	3.27+06
AM-241	W	8.54+04	3.45+03	1.49+05	7.12+05	1.57+06	1.21+07	4.55+06	5.69+07	8.54+04	3.27+06
AM-243	Y	9.25+04	3.81+03	1.57+05	7.12+05	1.57+06	1.21+07	4.55+06	6.05+07	9.25+04	3.37+06
AM-243	W	9.25+04	3.81+03	1.57+05	7.12+05	1.57+06	1.21+07	4.55+06	6.05+07	9.25+04	3.37+06
CM-242	Y	8.11+02	1.56+03	6.86+04	7.69+03	1.70+04	1.31+05	4.17+04	4.47+05	8.11+02	3.52+04
CM-242	W	8.11+02	1.56+03	6.86+04	7.69+03	1.70+04	1.31+05	4.17+04	4.47+05	8.11+02	3.52+04
CM-243	Y	2.47+04	1.76+03	7.45+04	2.12+05	4.47+05	3.58+06	1.25+06	1.43+07	2.45+04	8.70+05
CM-243	W	2.47+04	1.76+03	7.45+04	2.12+05	4.47+05	3.58+06	1.25+06	1.43+07	2.45+04	8.70+05
CM-244	Y	1.91+04	1.48+03	6.56+04	1.67+05	3.58+05	2.77+06	9.84+05	1.13+07	1.91+04	6.89+05
CM-244	W	1.91+04	1.48+03	6.56+04	1.67+05	3.58+05	2.77+06	9.84+05	1.13+07	1.91+04	6.89+05
CM-248	Y	1.73+05	2.32+04	3.28+05	1.19+06	2.41+06	1.88+07	1.82+06	2.62+07	1.43+05	2.59+06
CM-248	W	1.73+05	2.32+04	3.28+05	1.19+06	2.41+06	1.88+07	1.82+06	2.62+07	1.43+05	2.59+06
CF-252	Y	9.78+03	6.26+03	1.70+05	6.86+04	1.55+05	1.19+06	1.94+05	2.06+06	7.48+03	1.99+05
CF-252	W	9.78+03	6.26+03	1.70+05	6.86+04	1.55+05	1.19+06	1.94+05	2.06+06	7.48+03	1.99+05

TABLE D-23 . PDCF-5: PDCF for Direct Gamma Radiation (Volume Source)

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
C-14	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NA-22	D	1.26+07[#]	1.16+07	1.14+07	1.34+07	1.24+07	1.18+07	1.28+07	1.40+07	1.56+07	1.29+07
CL-36	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CL-36	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
FE-55	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
FE-55	Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CO-60	W	1.46+07	1.35+07	1.34+07	1.55+07	1.46+07	1.38+07	1.48+07	1.56+07	1.83+07	1.50+07
CO-60	Y	1.46+07	1.35+07	1.34+07	1.55+07	1.46+07	1.38+07	1.48+07	1.56+07	1.83+07	1.50+07
NI-59	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NI-59	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NI-63	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NI-63	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	D	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01
SR-90	Y	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01	1.92-01
NB-94	W	8.93+06	8.29+06	8.10+06	9.50+06	8.80+06	8.35+06	9.05+06	9.88+06	1.12+07	9.18+06
NB-94	Y	8.93+06	8.29+06	8.10+06	9.50+06	8.80+06	8.35+06	9.05+06	9.88+06	1.12+07	9.18+06
TC-99	D	1.24-02	1.08-02	1.02-02	1.39-02	1.15-02	1.12-02	9.12-03	2.17-02	1.83-02	1.28-02
TC-99	W	1.24-02	1.08-02	1.02-02	1.39-02	1.15-02	1.12-02	9.12-03	2.17-02	1.83-02	1.28-02
RU-106	Y	1.04+06	9.58+05	9.35+05	1.11+06	1.01+06	9.69+05	1.06+06	1.19+06	1.30+06	1.07+06
AG-108	D	8.19+06	7.58+06	7.41+06	8.81+06	8.02+06	7.69+06	8.36+06	9.43+06	1.03+07	8.47+06
AG-108	W	8.19+06	7.58+06	7.41+06	8.81+06	8.02+06	7.69+06	8.36+06	9.43+06	1.03+07	8.47+06
AG-108	Y	8.19+06	7.58+06	7.41+06	8.81+06	8.02+06	7.69+06	8.36+06	9.43+06	1.03+07	8.47+06
CD-109	D	1.00+05	7.20+04	6.89+04	2.27+05	1.17+05	6.32+04	4.98+04	1.70+05	1.79+05	1.52+05
CD-109	W	1.00+05	7.20+04	6.89+04	2.27+05	1.17+05	6.32+04	4.98+04	1.70+05	1.79+05	1.52+05
CD-109	Y	1.00+05	7.20+04	6.89+04	2.27+05	1.17+05	6.32+04	4.98+04	1.70+05	1.79+05	1.52+05
SN-126	D	1.11+07	1.02+07	9.97+06	1.18+07	1.08+07	1.03+07	1.11+07	1.28+07	1.39+07	1.14+07
SN-126	W	1.11+07	1.02+07	9.97+06	1.18+07	1.08+07	1.03+07	1.11+07	1.28+07	1.39+07	1.14+07
SB-125	D	2.28+06	2.09+06	2.04+06	2.49+06	2.25+06	2.12+06	2.32+06	2.70+06	2.88+06	2.38+06
SB-125	W	2.28+06	2.09+06	2.04+06	2.49+06	2.25+06	2.12+06	2.32+06	2.70+06	2.88+06	2.38+06
I-129	D	8.53+03	6.22+03	5.34+03	1.61+04	1.41+04	6.73+03	3.21+03	1.49+04	1.73+04	1.17+04
CS-134	D	8.84+06	8.14+06	7.95+06	9.41+06	8.65+06	8.27+06	8.96+06	9.92+06	1.11+07	9.09+06
CS-135	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CS-137	D	3.17+06	2.93+06	2.87+06	3.39+06	3.10+06	3.58+06	3.22+06	3.58+06	3.98+06	3.30+06
EU-152	W	5.80+06	5.32+06	5.25+06	6.20+06	5.76+06	5.43+06	5.78+06	6.62+06	7.39+06	5.96+06
EU-154	W	6.43+06	5.95+06	5.86+06	6.91+06	6.36+06	6.03+06	6.49+06	7.21+06	8.12+06	6.67+06
PB-210	W	5.81+03	4.49+03	3.90+03	1.03+04	6.57+03	4.97+03	2.61+03	1.06+04	9.74+03	7.53+03
AC-227	Y	1.61+06	1.46+06	1.42+06	1.78+06	1.55+06	1.48+06	1.54+06	2.24+06	2.14+06	1.68+06
AC-227	W	1.61+06	1.46+06	1.42+06	1.78+06	1.55+06	1.48+06	1.54+06	2.24+06	2.14+06	1.68+06
TH-228	Y	8.13+06	7.62+06	7.54+06	8.62+06	8.24+06	7.73+06	8.27+06	8.84+06	9.79+06	8.35+06
TH-228	W	8.13+06	7.62+06	7.54+06	8.62+06	8.24+06	7.73+06	8.27+06	8.84+06	9.79+06	8.35+06
TH-229	Y	1.12+06	1.03+06	1.01+06	1.20+06	1.10+06	1.05+06	1.11+06	1.30+06	1.40+06	1.15+06
TH-229	W	1.12+06	1.03+06	1.01+06	1.20+06	1.10+06	1.05+06	1.11+06	1.30+06	1.40+06	1.15+06
RN-222	*	8.96+06	8.24+06	8.18+06	9.51+06	8.84+06	8.42+06	9.09+06	9.99+06	9.99+06	9.15+06
RA-226	W	2.95+04	2.66+04	2.61+04	3.16+04	2.77+04	2.70+04	2.82+04	4.35+04	3.96+04	3.04+04
RA-228	W	4.03+06	3.73+06	3.68+06	4.31+06	3.97+06	3.77+06	4.06+06	4.51+06	5.09+06	4.15+06
TH-230	Y	5.25+02	4.21+02	4.05+02	1.07+03	4.44+02	4.25+02	3.53+02	8.98+02	7.47+02	7.48+02
TH-230	W	5.25+02	4.21+02	4.05+02	1.07+03	4.44+02	4.25+02	3.53+02	8.98+02	7.47+02	7.48+02
TH-232	Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
TH-232	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00

[#] Notation: 1.26+07 means 1.26 x 10^7

TABLE D-23 . PDCF-5: PDCF for Direct Gamma Radiation (Volume Source) (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	1.75+05	1.57+05	1.54+05	2.05+05	1.68+05	1.58+05	1.72+05	2.36+05	2.28+05	1.88+05
PA-231 W	1.75+05	1.57+05	1.54+05	2.05+05	1.68+05	1.58+05	1.72+05	2.36+05	2.28+05	1.88+05
U-232 Y	2.62+02	1.87+02	1.91+02	8.32+02	1.81+02	1.72+02	1.59+02	4.41+02	3.57+02	5.01+02
U-232 W	2.62+02	1.87+02	1.91+02	8.32+02	1.81+02	1.72+02	1.59+02	4.41+02	3.57+02	5.01+02
U-232 D	2.62+02	1.87+02	1.91+02	8.32+02	1.81+02	1.72+02	1.59+02	4.41+02	3.57+02	5.01+02
U-233 Y	1.63+02	1.34+02	1.33+02	3.16+02	1.31+02	1.30+02	1.27+02	2.68+02	2.23+02	2.26+02
U-233 W	1.63+02	1.34+02	1.33+02	3.16+02	1.31+02	1.30+02	1.27+02	2.68+02	2.23+02	2.26+02
U-233 D	1.63+02	1.34+02	1.33+02	3.16+02	1.31+02	1.30+02	1.27+02	2.68+02	2.23+02	2.26+02
U-234 Y	1.20+02	7.68+01	8.23+01	4.89+02	6.92+01	6.45+01	6.33+01	2.04+02	1.60+02	2.75+02
U-234 W	1.20+02	7.68+01	8.23+01	4.89+02	6.92+01	6.45+01	6.33+01	2.04+02	1.60+02	2.75+02
U-234 D	1.20+02	7.68+01	8.23+01	4.89+02	6.92+01	6.45+01	6.33+01	2.04+02	1.60+02	2.75+02
U-235 Y	6.58+05	5.92+05	5.79+05	7.46+05	6.19+05	6.01+05	6.19+05	9.92+05	8.95+05	6.98+05
U-235 W	6.58+05	5.92+05	5.79+05	7.46+05	6.19+05	6.01+05	6.19+05	9.92+05	8.95+05	6.98+05
U-235 D	6.58+05	5.92+05	5.79+05	7.46+05	6.19+05	6.01+05	6.19+05	9.92+05	8.95+05	6.98+05
U-236 Y	5.72+01	3.44+01	3.76+01	2.62+02	2.80+01	2.73+01	2.66+01	9.85+01	7.40+01	1.43+02
U-236 W	5.72+01	3.44+01	3.76+01	2.62+02	2.80+01	2.73+01	2.66+01	9.85+01	7.40+01	1.43+02
U-236 D	5.72+01	3.44+01	3.76+01	2.62+02	2.80+01	2.73+01	2.66+01	9.85+01	7.40+01	1.43+02
U-238 Y	6.13+04	5.66+04	5.56+04	6.55+04	6.02+04	5.74+04	6.13+04	7.01+04	7.75+04	6.30+04
U-238 W	6.13+04	5.66+04	5.56+04	6.55+04	6.02+04	5.74+04	6.13+04	7.01+04	7.75+04	6.30+04
U-238 D	6.13+04	5.66+04	5.56+04	6.55+04	6.02+04	5.74+04	6.13+04	7.01+04	7.75+04	6.30+04
NP-237 Y	9.87+05	8.87+05	8.65+05	1.10+06	9.37+05	9.01+05	9.46+05	1.38+06	1.30+06	1.03+06
NP-237 W	9.87+05	8.87+05	8.65+05	1.10+06	9.37+05	9.01+05	9.46+05	1.38+06	1.30+06	1.03+06
PU-236 Y	2.62+01	1.34+01	1.60+01	1.46+02	8.18+00	8.00+00	1.04+01	4.44+01	3.32+01	7.65+01
PU-236 W	2.62+01	1.34+01	1.60+01	1.46+02	8.18+00	8.00+00	1.04+01	4.44+01	3.32+01	7.65+01
PU-238 Y	1.74+01	8.05+00	1.03+01	1.09+02	3.80+00	3.74+00	6.42+00	2.92+01	2.15+01	5.61+01
PU-238 W	1.74+01	8.05+00	1.03+01	1.09+02	3.80+00	3.74+00	6.42+00	2.92+01	2.15+01	5.61+01
PU-239 Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-239 W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-240 Y	2.16+01	1.01+01	1.27+01	1.32+02	5.17+00	4.99+00	8.00+00	3.62+01	2.68+01	6.81+01
PU-240 W	2.16+01	1.01+01	1.27+01	1.32+02	5.17+00	4.99+00	8.00+00	3.62+01	2.68+01	6.81+01
PU-241 Y	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00
PU-241 W	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00	1.37+00
PU-242 Y	2.15+01	1.02+01	1.28+01	1.29+02	5.31+00	5.19+00	8.03+00	3.59+01	2.68+01	6.67+01
PU-242 W	2.15+01	1.02+01	1.28+01	1.29+02	5.31+00	5.19+00	8.03+00	3.59+01	2.68+01	6.67+01
PU-244 Y	1.49+06	1.37+06	1.35+06	1.62+06	1.46+06	1.39+06	1.52+06	1.70+06	1.87+06	1.55+06
PU-244 W	1.49+06	1.37+06	1.35+06	1.62+06	1.46+06	1.39+06	1.52+06	1.70+06	1.87+06	1.55+06
AM-241 Y	5.78+04	4.72+04	4.17+04	7.65+04	5.69+04	5.12+04	3.11+04	1.06+05	9.12+04	6.36+04
AM-241 W	5.78+04	4.72+04	4.17+04	7.65+04	5.69+04	5.12+04	3.11+04	1.06+05	9.12+04	6.36+04
AM-243 Y	8.31+05	7.32+05	7.04+05	9.41+05	7.77+05	7.53+05	7.04+05	1.31+06	1.16+06	8.67+05
AM-243 W	8.31+05	7.32+05	7.04+05	9.41+05	7.77+05	7.53+05	7.04+05	1.31+06	1.16+06	8.67+05
CM-242 Y	8.13+01	3.78+01	4.79+01	4.81+02	1.57+01	1.59+01	2.97+01	1.35+02	1.05+02	2.49+02
CM-242 W	8.13+01	3.78+01	4.79+01	4.81+02	1.57+01	1.59+01	2.97+01	1.35+02	1.05+02	2.49+02
CM-243 Y	4.96+05	4.45+05	4.32+05	5.55+05	4.62+05	4.49+05	4.62+05	7.37+05	6.69+05	5.21+05
CM-243 W	4.96+05	4.45+05	4.32+05	5.55+05	4.62+05	4.49+05	4.62+05	7.37+05	6.69+05	5.21+05
CM-244 Y	9.74+00	4.42+00	5.70+00	5.94+01	1.63+00	1.67+00	3.49+00	1.62+01	1.25+01	3.06+01
CM-244 W	9.74+00	4.42+00	5.70+00	5.94+01	1.63+00	1.67+00	3.49+00	1.62+01	1.25+01	3.06+01
CM-248 Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CM-248 W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CF-252 Y	2.52+03	1.25+03	1.51+03	1.36+04	5.68+02	5.59+02	9.49+02	4.18+03	3.46+03	7.13+03
CF-252 W	2.52+03	1.25+03	1.51+03	1.36+04	5.68+02	5.59+02	9.49+02	4.18+03	3.46+03	7.13+03

TABLE D-24 . PDCF-6: PDCF for Well Water Pathways

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	5.53+05[#]	7.14+05	9.46+05	5.49+05	5.66+05	5.48+05	5.46+05	4.34+05	5.48+05	5.94+05
C-14	*	7.73+06	1.10+07	1.33+07	1.75+07	9.65+06	1.12+07	3.08+07	6.42+07	8.09+06	1.40+07
NA-22	D	9.21+07	4.29+07	5.26+07	5.46+07	5.78+07	5.45+07	7.27+07	9.12+07	5.18+07	7.00+07
CL-36	D	3.72+07	5.16+07	3.72+07	3.84+07	3.72+07	3.72+07	3.72+07	3.72+07	3.72+07	3.80+07
CL-36	W	3.72+07	5.16+07	3.72+07	3.84+07	3.72+07	3.72+07	3.72+07	3.72+07	3.72+07	3.80+07
FE-55	W	9.68+05	2.51+04	9.65+05	4.78+05	8.78+05	1.66+06	8.74+05	5.96+05	8.62+05	1.06+06
FE-55	Y	9.68+05	2.51+04	9.65+05	4.78+05	8.78+05	1.66+06	8.74+05	5.96+05	8.62+05	1.06+06
CO-60	W	2.28+07	1.77+07	6.13+07	1.81+07	1.90+07	1.99+07	1.89+07	1.78+07	1.88+07	2.64+07
CO-60	Y	2.28+07	1.77+07	6.13+07	1.81+07	1.90+07	1.99+07	1.89+07	1.78+07	1.88+07	2.64+07
NI-59	D	1.44+05	1.69+05	1.11+06	2.35+05	1.46+05	1.46+05	1.50+05	1.49+05	1.60+05	2.34+05
NI-59	W	1.44+05	1.69+05	1.11+06	2.35+05	1.46+05	1.46+05	1.50+05	1.49+05	1.60+05	2.34+05
NI-63	D	3.49+05	4.31+05	3.77+06	6.40+05	3.49+05	3.49+05	3.49+05	3.49+05	3.49+05	6.40+05
NI-63	W	3.49+05	4.31+05	3.77+06	6.40+05	3.49+05	3.49+05	3.49+05	3.49+05	3.49+05	6.40+05
SR-90	D	6.78+06	9.92+05	8.81+07	1.07+08	6.78+06	6.46+06	4.87+08	9.74+08	6.78+06	9.91+07
SR-90	Y	3.74+05	9.92+05	1.09+08	5.68+06	3.40+05	3.24+05	2.43+07	4.87+07	3.40+05	1.28+07
NB-94	W	1.19+07	2.39+07	2.71+08	4.94+07	2.28+07	1.37+07	2.39+07	2.53+07	1.30+07	4.91+07
NB-94	Y	1.19+07	2.39+07	2.71+08	4.94+07	2.28+07	1.37+07	2.39+07	2.53+07	1.30+07	4.91+07
TC-99	D	7.74+05	2.27+06	7.81+06	5.22+05	1.12+06	1.53+06	7.86+05	1.00+06	3.44+07	2.49+06
TC-99	W	7.74+05	2.27+06	7.81+06	5.22+05	1.12+06	1.53+06	7.86+05	1.00+06	3.44+07	2.49+06
RU-106	Y	6.72+07	5.08+07	2.02+09	4.73+07	6.51+07	6.52+07	6.56+07	7.55+07	6.39+07	2.25+08
AG-108	D	1.22+07	1.42+07	5.35+07	2.16+07	1.48+07	4.92+07	1.28+07	1.21+07	1.17+07	2.12+07
AG-108	W	1.22+07	1.42+07	5.35+07	2.16+07	1.48+07	4.92+07	1.28+07	1.21+07	1.17+07	2.12+07
AG-108	Y	1.22+07	1.42+07	5.35+07	2.16+07	1.48+07	4.92+07	1.28+07	1.21+07	1.17+07	2.12+07
CD-109	D	1.55+06	1.97+06	2.20+07	2.35+07	1.94+08	3.51+07	1.78+06	1.63+06	1.39+06	1.69+07
CD-109	W	1.55+06	1.97+06	2.20+07	2.35+07	1.94+08	3.51+07	1.78+06	1.63+06	1.39+06	1.69+07
CD-109	Y	1.55+06	1.97+06	2.20+07	2.35+07	1.94+08	3.51+07	1.78+06	1.63+06	1.39+06	1.69+07
SN-126	D	1.61+07	2.42+07	3.95+08	5.84+07	1.77+07	1.61+07	3.51+07	5.75+07	1.84+07	5.80+07
SN-126	W	1.61+07	2.42+07	3.95+08	5.84+07	1.77+07	1.61+07	3.51+07	5.75+07	1.84+07	5.80+07
SB-125	D	2.52+06	3.31+06	2.59+07	5.62+06	2.71+06	3.14+06	3.24+06	5.12+06	3.06+06	5.51+06
SB-125	W	2.33+06	3.15+06	2.81+07	5.61+06	2.51+06	2.37+06	2.81+06	3.07+06	2.90+06	5.50+06
I-129	D	1.18+06	1.53+05	1.31+05	4.45+06	1.06+06	1.04+06	1.31+06	1.31+06	1.06+10	3.19+08
CS-134	D	2.36+08	7.86+07	8.93+07	1.06+08	1.50+08	1.50+08	1.40+08	1.35+08	1.22+08	1.68+08
CS-135	D	1.59+07	3.42+05	7.62+05	9.41+06	1.59+07	1.59+07	1.59+07	1.85+07	1.61+07	1.61+07
CS-137	D	1.45+08	3.38+07	3.96+07	7.31+07	1.13+08	1.15+08	1.08+08	1.17+08	9.95+07	1.20+08
EU-152	W	6.76+06	7.96+06	4.67+07	1.34+07	7.65+06	1.78+07	9.56+06	1.53+07	7.62+06	1.32+07
EU-154	W	7.18+06	9.21+06	8.04+07	1.75+07	8.08+06	2.13+07	1.11+07	2.56+07	8.18+06	1.72+07
PB-210	W	3.17+08	2.13+05	2.14+07	1.80+09	9.93+08	1.48+09	1.06+09	1.01+10	3.17+08	8.21+08
AC-227	Y	2.32+08	2.95+06	5.30+07	2.67+09	4.10+09	3.28+10	2.05+10	2.05+11	2.29+08	1.13+10
AC-227	W	2.28+08	2.95+06	5.30+07	2.67+09	4.10+09	3.28+10	2.05+10	2.05+11	2.29+08	1.13+10
TH-228	Y	2.85+07	2.18+07	5.08+08	5.86+07	2.63+07	4.14+07	4.15+08	4.31+09	2.92+07	2.38+08
TH-228	W	2.61+07	2.18+07	5.08+08	5.86+07	2.63+07	4.14+07	4.16+08	4.31+09	2.92+07	2.38+08
TH-229	Y	2.73+07	1.59+07	2.30+08	2.53+08	2.15+07	4.91+07	2.73+09	3.35+10	2.46+07	1.36+09
TH-229	W	2.28+07	1.59+07	2.30+08	2.54+08	2.15+07	4.92+07	2.74+09	3.35+10	2.46+07	1.36+09
RN-222	*	8.77+06	8.06+06	8.00+06	9.30+06	8.65+06	8.24+06	8.89+06	9.78+06	9.78+06	8.95+06
RA-226	W	6.31+08	5.78+06	3.53+08	3.64+09	6.31+08	6.31+08	2.35+09	2.14+10	6.31+08	1.48+09
RA-228	W	4.43+08	6.89+06	8.07+07	1.82+09	4.32+08	4.32+08	1.40+09	1.05+10	4.34+08	8.49+08
TH-230	Y	6.89+06	4.21+06	1.88+08	9.67+07	4.51+06	2.28+07	1.05+09	1.67+10	4.77+06	6.58+08
TH-230	W	4.99+06	4.21+06	1.88+08	9.68+07	4.52+06	2.28+07	1.05+09	1.68+10	4.78+06	6.59+08
TH-232	Y	5.98+06	3.63+06	1.57+08	1.01+08	3.92+06	1.97+07	1.15+09	1.88+10	4.12+06	7.24+08
TH-232	W	4.34+06	3.63+06	1.57+08	1.01+08	3.93+06	1.97+07	1.15+09	1.89+10	4.13+06	7.25+08

[#] Notation: 5.53+05 means 5.53 x 10^5

TABLE D-24 . PDCF-6: PDCF for Well Water Pathways (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	5.28+08	1.14+07	4.54+08	5.01+09	8.59+09	6.68+10	3.58+10	4.25+11	5.25+08	2.28+10
PA-231 W	5.25+08	1.14+07	4.54+08	5.01+09	8.59+09	6.68+10	3.58+10	4.25+11	5.25+08	2.28+10
U-232 Y	4.95+06	4.94+06	2.12+08	8.92+07	1.38+08	1.24+06	7.52+08	8.26+09	1.29+06	3.67+08
U-232 W	3.46+07	6.80+06	2.12+08	2.22+09	3.49+09	3.03+07	1.91+10	2.06+11	3.20+07	8.67+09
U-232 D	3.43+07	6.80+06	2.12+08	2.22+09	3.49+09	3.03+07	1.91+10	2.06+11	3.20+07	8.67+09
U-233 Y	2.92+06	4.43+06	1.91+08	2.55+07	7.18+07	6.43+05	1.02+07	1.48+08	6.78+05	2.72+07
U-233 W	1.86+07	5.83+06	1.80+08	6.14+08	1.80+09	1.60+07	2.54+08	3.71+09	1.69+07	2.76+08
U-233 D	1.84+07	5.83+06	1.80+08	6.14+08	1.80+09	1.60+07	2.54+08	3.71+09	1.69+07	2.76+08
U-234 Y	2.89+06	4.42+06	1.91+08	2.53+07	7.08+07	6.70+05	9.89+06	1.48+08	6.71+05	2.70+07
U-234 W	1.84+07	5.80+06	1.80+08	6.14+08	1.80+09	1.67+07	2.44+08	3.71+09	1.67+07	2.72+08
U-234 D	1.82+07	5.80+06	1.80+08	6.14+08	1.80+09	1.67+07	2.44+08	3.71+09	1.67+07	2.72+08
U-235 Y	3.63+06	5.09+06	2.02+08	2.39+07	6.49+07	1.46+06	8.65+06	1.28+08	1.82+06	2.72+07
U-235 W	1.82+07	6.49+06	1.91+08	5.52+08	1.59+09	1.54+07	1.91+08	3.07+09	1.66+07	2.38+08
U-235 D	1.80+07	6.49+06	1.91+08	5.52+08	1.59+09	1.54+07	1.91+08	3.07+09	1.66+07	2.38+08
U-236 Y	2.72+06	4.15+06	1.80+08	2.38+07	6.69+07	6.30+05	8.50+06	1.38+08	6.31+05	2.53+07
U-236 W	1.74+07	5.46+06	1.69+08	5.72+08	1.69+09	1.58+07	2.12+08	3.28+09	1.58+07	2.51+08
U-236 D	1.72+07	5.46+06	1.69+08	5.72+08	1.69+09	1.58+07	2.12+08	3.28+09	1.58+07	2.51+08
U-238 Y	1.28+07	1.34+07	1.89+08	3.35+07	7.32+07	1.02+07	1.84+07	1.28+08	1.36+07	3.47+07
U-238 W	2.66+07	1.46+07	1.79+08	5.51+08	1.60+09	2.38+07	2.11+08	2.98+09	2.79+07	2.43+08
U-238 D	2.65+07	1.46+07	1.79+08	5.51+08	1.60+09	2.38+07	2.11+08	2.98+09	2.79+07	2.43+08
NP-237 Y	1.34+08	6.84+06	2.18+08	1.09+09	2.28+09	1.74+10	6.62+09	9.44+10	1.32+08	5.09+09
NP-237 W	1.32+08	6.84+06	2.18+08	1.09+09	2.28+09	1.74+10	6.62+09	9.44+10	1.32+08	5.09+09
PU-236 Y	2.20+06	5.19+06	2.30+08	4.74+06	8.53+06	6.59+07	2.18+07	2.52+08	4.04+05	3.49+07
PU-236 W	6.60+05	5.19+06	2.30+08	4.75+06	8.63+06	6.67+07	2.21+07	2.55+08	4.09+05	3.49+07
PU-238 Y	5.83+06	4.93+06	2.20+08	2.98+07	6.02+07	4.63+08	1.79+08	2.21+09	3.40+06	1.44+08
PU-238 W	3.64+06	4.93+06	2.20+08	3.00+07	6.07+07	4.67+08	1.80+08	2.23+09	3.43+06	1.45+08
PU-239 Y	6.13+06	4.62+06	2.09+08	3.30+07	6.66+07	5.16+08	2.00+08	2.74+09	3.82+06	1.66+08
PU-239 W	4.04+06	4.62+06	2.09+08	3.32+07	6.72+07	5.20+08	2.02+08	2.76+09	3.86+06	1.66+08
PU-240 Y	6.12+06	4.62+06	2.09+08	3.30+07	6.65+07	5.16+08	2.00+08	2.74+09	3.81+06	1.66+08
PU-240 W	4.03+06	4.62+06	2.09+08	3.32+07	6.71+07	5.20+08	2.02+08	2.76+09	3.85+06	1.67+08
PU-241 Y	8.30+04	2.33+04	1.04+06	6.52+05	1.30+06	1.00+07	4.06+06	5.08+07	7.90+04	2.94+06
PU-241 W	7.86+04	2.33+04	1.04+06	6.57+05	1.31+06	1.01+07	4.10+06	5.13+07	7.95+04	2.96+06
PU-242 Y	5.82+06	4.40+06	1.99+08	3.14+07	6.32+07	4.95+08	1.89+08	2.74+09	3.63+06	1.61+08
PU-242 W	3.84+06	4.40+06	1.99+08	3.16+07	6.38+07	4.99+08	1.91+08	2.76+09	3.66+06	1.62+08
PU-244 Y	7.81+06	6.48+06	3.15+08	3.36+07	6.48+07	4.86+08	1.81+08	2.74+09	5.78+06	1.73+08
PU-244 W	5.83+06	6.48+06	3.15+08	3.38+07	6.54+07	4.90+08	1.82+08	2.76+09	5.81+06	1.73+08
AM-241 Y	1.36+08	5.47+06	2.33+08	1.11+09	2.44+09	1.88+10	7.10+09	8.87+10	1.33+08	5.09+09
AM-241 W	1.33+08	5.47+06	2.33+08	1.11+09	2.44+09	1.89+10	7.10+09	8.87+10	1.33+08	5.09+09
AM-243 Y	1.48+08	6.99+06	2.45+08	1.11+09	2.44+09	1.88+10	7.10+09	9.42+10	1.46+08	5.25+09
AM-243 W	1.46+08	6.99+06	2.45+08	1.11+09	2.44+09	1.89+10	7.10+09	9.43+10	1.46+08	5.25+09
CM-242 Y	3.53+06	5.48+06	2.40+08	2.70+07	5.97+07	4.60+08	1.46+08	1.57+09	2.84+06	1.23+08
CM-242 W	3.06+06	5.48+06	2.40+08	2.70+07	5.98+07	4.60+08	1.46+08	1.57+09	2.84+06	1.23+08
CM-243 Y	8.99+07	6.80+06	2.62+08	7.42+08	1.57+09	1.25+10	4.39+09	5.01+10	8.68+07	3.05+09
CM-243 W	8.76+07	6.80+06	2.62+08	7.43+08	1.57+09	1.25+10	4.39+09	5.01+10	8.68+07	3.05+09
CM-244 Y	6.93+07	5.18+06	2.30+08	5.85+08	1.25+09	9.71+09	3.45+09	3.97+10	6.68+07	2.41+09
CM-244 W	6.71+07	5.18+06	2.30+08	5.85+08	1.25+09	9.72+09	3.45+09	3.97+10	6.69+07	2.41+09
CM-248 Y	6.15+08	8.15+07	1.15+09	4.18+09	8.46+09	6.58+10	6.37+09	9.19+10	5.01+08	9.09+09
CM-248 W	6.07+08	8.15+07	1.15+09	4.18+09	8.46+09	6.58+10	6.37+09	9.19+10	5.01+08	9.09+09
CF-252 Y	3.78+07	2.19+07	5.95+08	2.40+08	5.43+08	4.18+09	6.79+08	7.21+09	2.62+07	6.96+08
CF-252 W	3.48+07	2.19+07	5.95+08	2.40+08	5.43+08	4.18+09	6.79+08	7.21+09	2.62+07	6.96+08

TABLE D-25 . PDCF-7: PDCF for Surface Water Pathways

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	5.54+05[#]	7.15+05	9.47+05	5.50+05	5.67+05	5.48+05	5.47+05	4.34+05	5.48+05	5.95+05
C-14	*	4.24+07	6.04+07	7.29+07	9.59+07	5.29+07	6.14+07	1.69+08	3.52+08	4.44+07	7.69+07
NA-22	D	1.15+08	5.22+07	6.46+07	6.67+07	7.11+07	6.69+07	9.01+07	1.13+08	6.26+07	8.66+07
CL-36	D	3.85+07	5.34+07	3.85+07	3.98+07	3.85+07	3.85+07	3.85+07	3.85+07	3.85+07	3.94+07
CL-36	W	3.85+07	5.34+07	3.85+07	3.98+07	3.85+07	3.85+07	3.85+07	3.85+07	3.85+07	3.94+07
FE-55	W	3.73+06	9.67+04	3.72+06	1.84+06	3.38+06	6.41+06	3.37+06	2.30+06	3.32+06	4.09+06
FE-55	Y	3.73+06	9.67+04	3.72+06	1.84+06	3.38+06	6.41+06	3.37+06	2.30+06	3.32+06	4.09+06
CO-60	W	2.75+07	2.05+07	8.32+07	2.05+07	2.21+07	2.36+07	2.18+07	2.00+07	2.05+07	3.26+07
CO-60	Y	2.75+07	2.05+07	8.32+07	2.05+07	2.21+07	2.36+07	2.18+07	2.00+07	2.05+07	3.26+07
NI-59	D	2.47+05	2.89+05	1.90+06	4.00+05	2.51+05	2.51+05	2.56+05	2.55+05	2.73+05	4.00+05
NI-59	W	2.47+05	2.89+05	1.90+06	4.00+05	2.51+05	2.51+05	2.56+05	2.55+05	2.73+05	4.00+05
NI-63	D	5.98+05	7.38+05	6.45+06	1.10+06	5.98+05	5.98+05	5.98+05	5.98+05	5.98+05	1.10+06
NI-63	W	5.98+05	7.38+05	6.45+06	1.10+06	5.98+05	5.98+05	5.98+05	5.98+05	5.98+05	1.10+06
SR-90	D	8.62+06	1.26+06	1.12+08	1.36+08	8.62+06	8.22+06	6.19+08	1.24+09	8.62+06	1.26+08
SR-90	Y	4.66+05	1.26+06	1.38+08	7.23+06	4.32+05	4.12+05	3.09+07	6.19+07	4.32+05	1.63+07
NB-94	W	1.44+08	6.14+08	9.86+09	1.53+09	5.55+08	2.29+08	5.89+08	6.11+08	1.07+08	1.53+09
NB-94	Y	1.44+08	6.14+08	9.86+09	1.53+09	5.55+08	2.29+08	5.89+08	6.11+08	1.07+08	1.53+09
TC-99	D	8.08+05	2.37+06	8.16+06	5.46+05	1.17+06	1.60+06	8.21+05	1.05+06	3.60+07	2.60+06
TC-99	W	8.09+05	2.37+06	8.16+06	5.46+05	1.17+06	1.60+06	8.21+05	1.05+06	3.60+07	2.60+06
RU-106	Y	7.03+07	5.31+07	2.11+09	4.95+07	6.81+07	6.83+07	6.87+07	7.91+07	6.69+07	2.35+08
AG-108	D	1.40+07	1.72+07	7.59+07	2.76+07	1.79+07	6.92+07	1.47+07	1.31+07	1.21+07	2.72+07
AG-108	W	1.40+07	1.72+07	7.59+07	2.76+07	1.79+07	6.92+07	1.47+07	1.31+07	1.21+07	2.72+07
AG-108	Y	1.40+07	1.72+07	7.59+07	2.76+07	1.79+07	6.92+07	1.47+07	1.31+07	1.21+07	2.72+07
CD-109	D	5.50+06	7.08+06	7.98+07	8.50+07	7.05+08	1.27+08	6.41+06	5.72+06	4.84+06	6.12+07
CD-109	W	5.50+06	7.08+06	7.98+07	8.50+07	7.05+08	1.27+08	6.41+06	5.72+06	4.84+06	6.12+07
CD-109	Y	5.51+06	7.08+06	7.98+07	8.50+07	7.05+08	1.27+08	6.41+06	5.72+06	4.84+06	6.12+07
SN-126	D	6.43+07	1.53+08	3.87+09	4.82+08	8.35+07	7.10+07	2.54+08	4.63+08	6.26+07	4.81+08
SN-126	W	6.43+07	1.53+08	3.87+09	4.82+08	8.35+07	7.10+07	2.54+08	4.63+08	6.26+07	4.81+08
SB-125	D	2.52+06	3.33+06	2.63+07	5.67+06	2.71+06	3.16+06	3.26+06	5.16+06	3.07+06	5.56+06
SB-125	W	2.33+06	3.16+06	2.84+07	5.66+06	2.52+06	2.38+06	2.82+06	3.08+06	2.90+06	5.54+06
I-129	D	1.27+06	1.62+05	1.38+05	4.80+06	1.14+06	1.12+06	1.41+06	1.40+06	1.15+10	3.44+08
CS-134	D	2.46+09	7.72+08	8.89+08	1.06+09	1.54+09	1.54+09	1.43+09	1.37+09	1.21+09	1.72+09
CS-135	D	1.72+08	3.68+06	8.20+06	1.01+08	1.72+08	1.72+08	1.72+08	1.99+08	1.73+08	1.73+08
CS-137	D	1.54+09	3.37+08	4.00+08	7.56+08	1.19+09	1.21+09	1.13+09	1.23+09	1.03+09	1.26+09
EU-152	W	7.80+06	1.07+07	9.01+07	2.10+07	9.67+06	3.09+07	1.35+07	2.43+07	7.91+06	2.08+07
EU-154	W	8.12+06	1.28+07	1.59+08	2.87+07	1.00+07	3.73+07	1.61+07	4.49+07	8.42+06	2.84+07
PB-210	W	5.54+08	3.67+05	3.75+07	3.14+09	1.74+09	2.58+09	1.85+09	1.77+10	5.54+08	1.43+09
AC-227	Y	3.61+08	3.39+06	8.21+07	4.19+09	6.45+09	5.16+10	3.22+10	3.22+11	3.58+08	1.78+10
AC-227	W	3.57+08	3.39+06	8.20+07	4.19+09	6.45+09	5.16+10	3.22+10	3.22+11	3.58+08	1.78+10
TH-228	Y	3.38+07	2.53+07	8.40+08	8.55+07	3.18+07	5.79+07	6.84+08	7.20+09	3.44+07	3.87+08
TH-228	W	3.14+07	2.53+07	8.40+08	8.55+07	3.18+07	5.79+07	6.85+08	7.21+09	3.44+07	3.87+08
TH-229	Y	3.44+07	1.93+07	3.79+08	4.16+08	2.82+07	7.49+07	4.57+09	5.61+10	3.12+07	2.27+09
TH-229	W	3.00+07	1.93+07	3.79+08	4.16+08	2.82+07	7.49+07	4.57+09	5.62+10	3.12+07	2.28+09
RN-222	*	8.77+06	8.06+06	8.00+06	9.30+06	8.65+06	8.24+06	8.89+06	9.78+06	9.78+06	8.95+06
RA-226	W	9.82+08	8.97+06	5.49+08	5.66+09	9.82+08	9.82+08	3.66+09	3.33+10	9.82+08	2.30+09
RA-228	W	6.87+08	8.29+06	1.23+08	2.83+09	6.70+08	6.70+08	2.17+09	1.63+10	6.72+08	1.32+09
TH-230	Y	1.01+07	7.06+06	3.15+08	1.62+08	7.56+06	3.82+07	1.75+09	2.81+10	8.00+06	1.10+09
TH-230	W	8.21+06	7.06+06	3.15+08	1.62+08	7.56+06	3.83+07	1.75+09	2.81+10	8.00+06	1.10+09
TH-232	Y	8.79+06	6.08+06	2.63+08	1.69+08	6.58+06	3.30+07	1.93+09	3.16+10	6.91+06	1.21+09
TH-232	W	7.15+06	6.08+06	2.63+08	1.69+08	6.58+06	3.30+07	1.93+09	3.16+10	6.91+06	1.21+09

[#] Notation: 5.54+05 means 5.54 x 10^5

TABLE D-25 . PDCF-7: PDCF for Surface Water Pathways (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	5.68+08	1.22+07	4.89+08	5.40+09	9.26+09	7.20+10	3.86+10	4.58+11	5.66+08	2.46+10
PA-231 W	5.66+08	1.22+07	4.89+08	5.40+09	9.26+09	7.20+10	3.86+10	4.58+11	5.66+08	2.46+10
U-232 Y	5.05+06	5.28+06	2.27+08	9.53+07	1.47+08	1.32+06	8.04+08	8.84+09	1.38+06	3.92+08
U-232 W	3.70+07	7.27+06	2.27+08	2.38+09	3.74+09	3.24+07	2.04+10	2.21+11	3.42+07	9.28+09
U-232 D	3.67+07	7.27+06	2.27+08	2.38+09	3.74+09	3.24+07	2.04+10	2.21+11	3.42+07	9.28+09
U-233 Y	2.97+06	4.74+06	2.04+08	2.73+07	7.68+07	6.87+05	1.09+07	1.59+08	7.26+05	2.91+07
U-233 W	1.99+07	6.24+06	1.93+08	6.57+08	1.93+09	1.71+07	2.72+08	3.96+09	1.81+07	2.96+08
U-233 D	1.97+07	6.24+06	1.93+08	6.57+08	1.93+09	1.71+07	2.72+08	3.96+09	1.81+07	2.96+08
U-234 Y	2.94+06	4.72+06	2.04+08	2.70+07	7.58+07	7.16+05	1.06+07	1.59+08	7.17+05	2.89+07
U-234 W	1.97+07	6.21+06	1.93+08	6.57+08	1.93+09	1.79+07	2.60+08	3.96+09	1.79+07	2.91+08
U-234 D	1.95+07	6.21+06	1.93+08	6.57+08	1.93+09	1.79+07	2.60+08	3.96+09	1.79+07	2.91+08
U-235 Y	3.68+06	5.39+06	2.16+08	2.55+07	6.94+07	1.50+06	9.19+06	1.37+08	1.87+06	2.90+07
U-235 W	1.94+07	6.88+06	2.05+08	5.90+08	1.70+09	1.64+07	2.05+08	3.29+09	1.76+07	2.55+08
U-235 D	1.92+07	6.88+06	2.05+08	5.90+08	1.70+09	1.64+07	2.05+08	3.29+09	1.76+07	2.55+08
U-236 Y	2.77+06	4.44+06	1.93+08	2.54+07	7.16+07	6.74+05	9.10+06	1.47+08	6.75+05	2.70+07
U-236 W	1.86+07	5.84+06	1.81+08	6.12+08	1.81+09	1.69+07	2.27+08	3.51+09	1.69+07	2.68+08
U-236 D	1.84+07	5.84+06	1.81+08	6.12+08	1.81+09	1.69+07	2.27+08	3.51+09	1.69+07	2.68+08
U-238 Y	1.29+07	1.36+07	2.02+08	3.50+07	7.76+07	1.02+07	1.89+07	1.36+08	1.36+07	3.63+07
U-238 W	2.78+07	1.50+07	1.91+08	5.89+08	1.71+09	2.48+07	2.25+08	3.18+09	2.89+07	2.60+08
U-238 D	2.76+07	1.50+07	1.91+08	5.89+08	1.71+09	2.48+07	2.25+08	3.18+09	2.89+07	2.60+08
NP-237 Y	1.90+08	9.30+06	3.12+08	1.56+09	3.26+09	2.49+10	9.48+09	1.35+11	1.88+08	7.29+09
NP-237 W	1.88+08	9.30+06	3.12+08	1.56+09	3.26+09	2.49+10	9.48+09	1.35+11	1.88+08	7.29+09
PU-236 Y	2.30+06	6.54+06	2.90+08	5.96+06	1.07+07	8.30+07	2.75+07	3.17+08	5.09+05	4.40+07
PU-236 W	7.65+05	6.54+06	2.90+08	5.97+06	1.08+07	8.38+07	2.77+07	3.20+08	5.13+05	4.39+07
PU-238 Y	6.71+06	6.21+06	2.77+08	3.76+07	7.58+07	5.83+08	2.25+08	2.78+09	4.28+06	1.82+08
PU-238 W	4.52+06	6.21+06	2.77+08	3.78+07	7.63+07	5.87+08	2.27+08	2.80+09	4.31+06	1.83+08
PU-239 Y	7.12+06	5.83+06	2.64+08	4.15+07	8.39+07	6.49+08	2.52+08	3.44+09	4.81+06	2.08+08
PU-239 W	5.03+06	5.83+06	2.64+08	4.18+07	8.45+07	6.54+08	2.54+08	3.47+09	4.85+06	2.09+08
PU-240 Y	7.11+06	5.83+06	2.64+08	4.15+07	8.37+07	6.49+08	2.52+08	3.44+09	4.80+06	2.08+08
PU-240 W	5.01+06	5.83+06	2.64+08	4.18+07	8.43+07	6.54+08	2.53+08	3.47+09	4.83+06	2.09+08
PU-241 Y	1.03+05	2.94+04	1.31+06	8.21+05	1.63+06	1.26+07	5.12+06	6.40+07	9.94+04	3.70+06
PU-241 W	9.91+04	2.94+04	1.31+06	8.26+05	1.64+06	1.27+07	5.15+06	6.44+07	9.99+04	3.72+06
PU-242 Y	6.76+06	5.55+06	2.50+08	3.95+07	7.96+07	6.23+08	2.38+08	3.44+09	4.57+06	2.03+08
PU-242 W	4.78+06	5.55+06	2.50+08	3.98+07	8.02+07	6.27+08	2.40+08	3.47+09	4.60+06	2.04+08
PU-244 Y	8.82+06	7.76+06	3.97+08	4.18+07	8.12+07	6.11+08	2.27+08	3.45+09	6.71+06	2.17+08
PU-244 W	6.84+06	7.76+06	3.97+08	4.20+07	8.17+07	6.15+08	2.29+08	3.47+09	6.75+06	2.18+08
AM-241 Y	4.63+08	1.87+07	8.05+08	3.83+09	8.43+09	6.52+10	2.45+10	3.07+11	4.60+08	1.76+10
AM-241 W	4.60+08	1.87+07	8.05+08	3.83+09	8.43+09	6.52+10	2.45+10	3.07+11	4.60+08	1.76+10
AM-243 Y	5.02+08	2.16+07	8.44+08	3.84+09	8.44+09	6.52+10	2.45+10	3.26+11	5.00+08	1.81+10
AM-243 W	5.00+08	2.16+07	8.44+08	3.84+09	8.44+09	6.52+10	2.45+10	3.26+11	5.00+08	1.81+10
CM-242 Y	6.72+06	1.16+07	5.10+08	5.72+07	1.27+08	9.75+08	3.10+08	3.33+09	6.03+06	2.62+08
CM-242 W	6.25+06	1.16+07	5.10+08	5.72+07	1.27+08	9.76+08	3.10+08	3.33+09	6.03+06	2.62+08
CM-243 Y	1.87+08	1.37+07	5.55+08	1.57+09	3.33+09	2.66+10	9.31+09	1.06+11	1.83+08	6.47+09
CM-243 W	1.85+08	1.37+07	5.55+08	1.58+09	3.33+09	2.66+10	9.31+09	1.06+11	1.83+08	6.48+09
CM-244 Y	1.44+08	1.10+07	4.88+08	1.24+09	2.66+09	2.06+10	7.32+09	8.42+10	1.42+08	5.12+09
CM-244 W	1.42+08	1.10+07	4.88+08	1.24+09	2.66+09	2.06+10	7.32+09	8.43+10	1.42+08	5.12+09
CM-248 Y	1.29+09	1.73+08	2.44+09	8.87+09	1.80+10	1.40+11	1.35+10	1.95+11	1.06+09	1.93+10
CM-248 W	1.29+09	1.73+08	2.44+09	8.87+09	1.80+10	1.40+11	1.35+10	1.95+11	1.06+09	1.93+10
CF-252 Y	7.62+07	4.66+07	1.26+09	5.10+08	1.15+09	8.87+09	1.44+09	1.53+10	5.56+07	1.48+09
CF-252 W	7.32+07	4.66+07	1.26+09	5.10+08	1.15+09	8.87+09	1.44+09	1.53+10	5.57+07	1.48+09

TABLE D-26 . PDCF-8: PDCF for Area Exposure Scenario

NUCLIDE		LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
H-3	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
C-14	*	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NA-22	D	1.85+08[#]	1.70+08	1.68+08	1.97+08	1.82+08	1.73+08	1.88+08	2.06+08	2.30+08	1.90+08
CL-36	D	2.65-01	6.85-02	1.38-01	3.36+00	4.03-04	3.26-03	8.70-02	4.29-01	1.68-01	1.62+00
CL-36	W	2.65-01	6.85-02	1.38-01	3.36+00	4.03-04	3.26-03	8.70-02	4.29-01	1.68-01	1.62+00
FE-55	W	1.26+03	3.24+02	6.56+02	1.59+04	1.91+00	1.55+01	4.14+02	2.04+03	7.96+02	7.66+03
FE-55	Y	1.26+03	3.24+02	6.56+02	1.59+04	1.91+00	1.55+01	4.14+02	2.04+03	7.96+02	7.66+03
CO-60	W	2.02+08	1.86+08	1.85+08	2.14+08	2.01+08	1.91+08	2.04+08	2.16+08	2.52+08	2.07+08
CO-60	Y	2.02+08	1.86+08	1.85+08	2.14+08	2.01+08	1.91+08	2.04+08	2.16+08	2.52+08	2.07+08
NI-59	D	2.37+03	6.11+02	1.23+03	3.00+04	3.59+00	2.91+01	7.77+02	3.85+03	1.50+03	1.45+04
NI-59	W	2.37+03	6.11+02	1.23+03	3.00+04	3.59+00	2.91+01	7.77+02	3.85+03	1.50+03	1.45+04
NI-63	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NI-63	W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
SR-90	Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
NB-94	W	1.40+08	1.30+08	1.27+08	1.49+08	1.38+08	1.31+08	1.42+08	1.55+08	1.76+08	1.44+08
NB-94	Y	1.40+08	1.30+08	1.27+08	1.49+08	1.38+08	1.31+08	1.42+08	1.55+08	1.76+08	1.44+08
TC-99	D	4.93+01	4.30+01	4.04+01	5.52+01	4.56+01	4.44+01	3.62+01	8.63+01	7.26+01	5.08+01
TC-99	W	4.93+01	4.30+01	4.04+01	5.52+01	4.56+01	4.44+01	3.62+01	8.63+01	7.26+01	5.08+01
RU-106	Y	1.84+07	1.70+07	1.66+07	1.97+07	1.80+07	1.72+07	1.88+07	2.11+07	2.30+07	1.90+07
AG-108	D	1.46+08	1.35+08	1.32+08	1.57+08	1.43+08	1.37+08	1.49+08	1.68+08	1.84+08	1.51+08
AG-108	W	1.46+08	1.35+08	1.32+08	1.57+08	1.43+08	1.37+08	1.49+08	1.68+08	1.84+08	1.51+08
AG-108	Y	1.46+08	1.35+08	1.32+08	1.57+08	1.43+08	1.37+08	1.49+08	1.68+08	1.84+08	1.51+08
CD-109	D	7.75+05	5.58+05	5.34+05	1.76+06	9.07+05	4.90+05	3.86+05	1.32+06	1.39+06	1.18+06
CD-109	W	7.75+05	5.58+05	5.34+05	1.76+06	9.07+05	4.90+05	3.86+05	1.32+06	1.39+06	1.18+06
CD-109	Y	7.75+05	5.58+05	5.34+05	1.76+06	9.07+05	4.90+05	3.86+05	1.32+06	1.39+06	1.18+06
SN-126	D	1.81+08	1.67+08	1.63+08	1.93+08	1.76+08	1.68+08	1.82+08	2.09+08	2.27+08	1.86+08
SN-126	W	1.81+08	1.67+08	1.63+08	1.93+08	1.76+08	1.68+08	1.82+08	2.09+08	2.27+08	1.86+08
SB-125	D	3.83+07	3.52+07	3.44+07	4.19+07	3.78+07	3.57+07	3.90+07	4.55+07	4.85+07	4.00+07
SB-125	W	3.83+07	3.52+07	3.44+07	4.19+07	3.78+07	3.57+07	3.90+07	4.55+07	4.85+07	4.00+07
I-129	D	1.07+06	7.81+05	6.70+05	2.02+06	1.77+06	8.44+05	4.03+05	1.87+06	2.17+06	1.47+06
CS-134	D	1.39+08	1.28+08	1.25+08	1.48+08	1.36+08	1.30+08	1.41+08	1.56+08	1.74+08	1.43+08
CS-135	D	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
CS-137	D	5.08+07	4.69+07	4.59+07	5.43+07	4.97+07	5.73+07	5.15+07	5.74+07	6.37+07	5.29+07
EU-152	W	9.73+07	8.92+07	8.81+07	1.04+08	9.66+07	9.10+07	9.69+07	1.11+08	1.24+08	1.00+08
EU-154	W	1.06+08	9.81+07	9.66+07	1.14+08	1.05+08	9.95+07	1.07+08	1.19+08	1.34+08	1.10+08
PB-210	W	1.45+05	1.12+05	9.73+04	2.57+05	1.64+05	1.24+05	6.52+04	2.64+05	2.43+05	1.88+05
AC-227	Y	4.09+07	3.69+07	3.59+07	4.51+07	3.93+07	3.76+07	3.89+07	5.68+07	5.43+07	4.26+07
AC-227	W	4.09+07	3.69+07	3.59+07	4.51+07	3.93+07	3.76+07	3.89+07	5.68+07	5.43+07	4.26+07
TH-228	Y	3.00+08	2.81+08	2.78+08	3.18+08	3.04+08	2.85+08	3.05+08	3.26+08	3.61+08	3.08+08
TH-228	W	3.00+08	2.81+08	2.78+08	3.18+08	3.04+08	2.85+08	3.05+08	3.26+08	3.61+08	3.08+08
TH-229	Y	1.99+08	1.82+08	1.80+08	2.13+08	1.96+08	1.86+08	1.97+08	2.30+08	2.49+08	2.05+08
TH-229	W	1.99+08	1.82+08	1.80+08	2.13+08	1.96+08	1.86+08	1.97+08	2.30+08	2.49+08	2.05+08
RN-222	*	1.48+08	1.36+08	1.35+08	1.57+08	1.46+08	1.39+08	1.50+08	1.65+08	1.65+08	1.51+08
RA-226	W	6.22+05	5.62+05	5.51+05	6.67+05	5.85+05	5.70+05	5.96+05	9.18+05	8.36+05	6.42+05
RA-228	W	8.00+07	7.41+07	7.30+07	8.56+07	7.89+07	7.48+07	8.07+07	8.96+07	1.01+08	8.24+07
TH-230	Y	3.85+04	3.09+04	2.97+04	7.85+04	3.26+04	3.12+04	2.59+04	6.59+04	5.48+04	5.49+04
TH-230	W	3.85+04	3.09+04	2.97+04	7.85+04	3.26+04	3.12+04	2.59+04	6.59+04	5.48+04	5.49+04
TH-232	Y	1.96+04	1.43+04	1.41+04	5.74+04	1.48+04	1.38+04	1.13+04	3.41+04	2.74+04	3.54+04
TH-232	W	1.96+04	1.43+04	1.41+04	5.74+04	1.48+04	1.38+04	1.13+04	3.41+04	2.74+04	3.54+04

[#] Notation: 1.85+08 means 1.85 x 10^8

TABLE D-26 . PDCF-8: PDCF for Area Exposure Scenario (continued)

NUCLIDE	LUNGS	S. WALL	LI WALL	T. BODY	KIDNEYS	LIVER	RED MAR	BONE	THYROID	ICRP
PA-231 Y	2.77+06	2.49+06	2.44+06	3.25+06	2.66+06	2.51+06	2.73+06	3.74+06	3.62+06	2.98+06
PA-231 W	2.77+06	2.49+06	2.44+06	3.25+06	2.66+06	2.51+06	2.73+06	3.74+06	3.62+06	2.98+06
U-232 Y	2.86+04	2.04+04	2.08+04	9.07+04	1.97+04	1.87+04	1.73+04	4.81+04	3.89+04	5.46+04
U-232 W	2.86+04	2.04+04	2.08+04	9.07+04	1.97+04	1.87+04	1.73+04	4.81+04	3.89+04	5.46+04
U-232 D	2.86+04	2.04+04	2.08+04	9.07+04	1.97+04	1.87+04	1.73+04	4.81+04	3.89+04	5.46+04
U-233 Y	2.27+04	1.87+04	1.86+04	4.41+04	1.83+04	1.81+04	1.77+04	3.74+04	3.11+04	3.16+04
U-233 W	2.27+04	1.87+04	1.86+04	4.41+04	1.83+04	1.81+04	1.77+04	3.74+04	3.11+04	3.16+04
U-233 D	2.27+04	1.87+04	1.86+04	4.41+04	1.83+04	1.81+04	1.77+04	3.74+04	3.11+04	3.16+04
U-234 Y	1.74+04	1.11+04	1.19+04	7.07+04	1.00+04	9.33+03	9.15+03	2.95+04	2.31+04	3.98+04
U-234 W	1.74+04	1.11+04	1.19+04	7.07+04	1.00+04	9.33+03	9.15+03	2.95+04	2.31+04	3.98+04
U-234 D	1.74+04	1.11+04	1.19+04	7.07+04	1.00+04	9.33+03	9.15+03	2.95+04	2.31+04	3.98+04
U-235 Y	1.50+07	1.35+07	1.32+07	1.70+07	1.41+07	1.37+07	1.41+07	2.26+07	2.04+07	1.59+07
U-235 W	1.50+07	1.35+07	1.32+07	1.70+07	1.41+07	1.37+07	1.41+07	2.26+07	2.04+07	1.59+07
U-235 D	1.50+07	1.35+07	1.32+07	1.70+07	1.41+07	1.37+07	1.41+07	2.26+07	2.04+07	1.59+07
U-236 Y	1.40+04	8.41+03	9.19+03	6.41+04	6.85+03	6.67+03	6.52+03	2.41+04	1.81+04	3.51+04
U-236 W	1.40+04	8.41+03	9.19+03	6.41+04	6.85+03	6.67+03	6.52+03	2.41+04	1.81+04	3.51+04
U-236 D	1.40+04	8.41+03	9.19+03	6.41+04	6.85+03	6.67+03	6.52+03	2.41+04	1.81+04	3.51+04
U-238 Y	1.73+08	1.60+08	1.57+08	1.85+08	1.70+08	1.62+08	1.73+08	1.98+08	2.19+08	1.78+08
U-238 W	1.73+08	1.60+08	1.57+08	1.85+08	1.70+08	1.62+08	1.73+08	1.98+08	2.19+08	1.78+08
U-238 D	1.73+08	1.60+08	1.57+08	1.85+08	1.70+08	1.62+08	1.73+08	1.98+08	2.19+08	1.78+08
NP-237 Y	2.18+07	1.96+07	1.91+07	2.43+07	2.07+07	1.99+07	2.09+07	3.04+07	2.88+07	2.28+07
NP-237 W	2.18+07	1.96+07	1.91+07	2.43+07	2.07+07	1.99+07	2.09+07	3.04+07	2.88+07	2.28+07
PU-236 Y	1.59+04	8.15+03	9.70+03	8.85+04	4.96+03	4.85+03	6.33+03	2.69+04	2.01+04	4.64+04
PU-236 W	1.59+04	8.15+03	9.70+03	8.85+04	4.96+03	4.85+03	6.33+03	2.69+04	2.01+04	4.64+04
PU-238 Y	1.22+04	5.66+03	7.22+03	7.66+04	2.67+03	2.63+03	4.51+03	2.05+04	1.51+04	3.94+04
PU-238 W	1.22+04	5.66+03	7.22+03	7.66+04	2.67+03	2.63+03	4.51+03	2.05+04	1.51+04	3.94+04
PU-239 Y	8.95+03	6.03+03	6.55+03	3.37+04	4.85+03	4.85+03	5.59+03	1.47+04	1.17+04	1.94+04
PU-239 W	8.95+03	6.03+03	6.55+03	3.37+04	4.85+03	4.85+03	5.59+03	1.47+04	1.17+04	1.94+04
PU-240 Y	1.20+04	5.63+03	7.07+03	7.33+04	2.87+03	2.77+03	4.44+03	2.01+04	1.49+04	3.78+04
PU-240 W	1.20+04	5.63+03	7.07+03	7.33+04	2.87+03	2.77+03	4.44+03	2.01+04	1.49+04	3.78+04
PU-241 Y	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-241 W	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00	.00+00
PU-242 Y	1.01+04	4.81+03	6.00+03	6.07+04	2.50+03	2.44+03	3.78+03	1.69+04	1.26+04	3.14+04
PU-242 W	1.01+04	4.81+03	6.00+03	6.07+04	2.50+03	2.44+03	3.78+03	1.69+04	1.26+04	3.14+04
PU-244 Y	2.91+07	2.68+07	2.63+07	3.16+07	2.85+07	2.72+07	2.96+07	3.32+07	3.64+07	3.02+07
PU-244 W	2.91+07	2.68+07	2.63+07	3.16+07	2.85+07	2.72+07	2.96+07	3.32+07	3.64+07	3.02+07
AM-241 Y	2.01+06	1.64+06	1.45+06	2.66+06	1.98+06	1.78+06	1.08+06	3.69+06	3.17+06	2.21+06
AM-241 W	2.01+06	1.64+06	1.45+06	2.66+06	1.98+06	1.78+06	1.08+06	3.69+06	3.17+06	2.21+06
AM-243 Y	2.03+07	1.79+07	1.72+07	2.30+07	1.90+07	1.84+07	1.72+07	3.20+07	2.84+07	2.12+07
AM-243 W	2.03+07	1.79+07	1.72+07	2.30+07	1.90+07	1.84+07	1.72+07	3.20+07	2.84+07	2.12+07
CM-242 Y	1.44+04	6.70+03	8.48+03	8.52+04	2.78+03	2.82+03	5.26+03	2.39+04	1.86+04	4.41+04
CM-242 W	1.44+04	6.70+03	8.48+03	8.52+04	2.78+03	2.82+03	5.26+03	2.39+04	1.86+04	4.41+04
CM-243 Y	1.17+07	1.05+07	1.02+07	1.31+07	1.09+07	1.06+07	1.09+07	1.74+07	1.58+07	1.23+07
CM-243 W	1.17+07	1.05+07	1.02+07	1.31+07	1.09+07	1.06+07	1.09+07	1.74+07	1.58+07	1.23+07
CM-244 Y	1.24+04	5.63+03	7.26+03	7.56+04	2.07+03	2.12+03	4.44+03	2.06+04	1.59+04	3.89+04
CM-244 W	1.24+04	5.63+03	7.26+03	7.56+04	2.07+03	2.12+03	4.44+03	2.06+04	1.59+04	3.89+04
CM-248 Y	9.04+03	4.19+03	5.30+03	5.37+04	1.76+03	1.75+03	3.27+03	1.50+04	1.17+04	2.78+04
CM-248 W	9.04+03	4.19+03	5.30+03	5.37+04	1.76+03	1.75+03	3.27+03	1.50+04	1.17+04	2.78+04
CF-252 Y	1.09+04	5.41+03	6.56+03	5.89+04	2.46+03	2.42+03	4.11+03	1.81+04	1.50+04	3.09+04
CF-252 W	1.09+04	5.41+03	6.56+03	5.89+04	2.46+03	2.42+03	4.11+03	1.81+04	1.50+04	3.09+04

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APPENDIX E

SPECIAL SCENARIOS

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APPENDIX E : Special Scenarios

This appendix addresses some special cases associated with release/transport/pathway mechanisms (scenarios) including water well drilling and parameters, plus metal corrosion and corrosion rates. It also presents data on these cases that are used in the release scenarios.

A brief introduction to the appendix is presented in Chapter E.1 followed by a discussion of water well drilling in Chapter E.2. Corrosion of activated metals is discussed in Chapter E.3.

E.1 INTRODUCTION

This report addresses radioactive wastes having concentrations that exceed Part 61 Class C limits. According to Part 61, these wastes are "not generally acceptable for near-surface disposal." However, Part 61 also specifies that:

> "Waste that is not generally acceptable for near-surface disposal is waste for which waste form and disposal methods must be different, and in general more stringent, than those specified for Class C waste. In the absence of specific requirements in this part, proposals for disposal of this waste may be submitted to the Commission for approval, pursuant to 61.58 of this part."

In other words, these wastes are to be considered on a "case-by-case" basis. Consequently, it appears that the following special cases and scenarios must be addressed:

- Information on available water well drilling methodologies is needed as a foundation for the formulation of a comprehensive scenario involving direct contact with wastes brought to the surface;
- Interaction of activated metals with the environment (i.e., transfer agents such as wind or water) must be quantified with special attention on their corrosion characteristics and self-shielding properties;

These subjects are addressed in the following sections.

E.2 WATER WELL DRILLING

This chapter discusses and presents data on water well drilling methods and parameters.

E.2.1 Background

One of the most direct ways of accessing disposed radioactivity is through the hypothetical drilling of a well by an inadvertant intruder. This scenario was not considered in the original Part 61 analysis methodology

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since other, more conservative, direct access scenarios (intruder-agriculture and intruder-construction) were considered which were judged adequate for the purpose of establishing licensing requirements for near-surface waste disposal. However, consumption and use of water from an intruder well was used as part of estimating residual impacts from waste disposal.

Since then, it appears to be reasonable to include well drilling as a distinct direct access scenario, and to consider the material brought to the surface as part of the intruder-agriculture scenario. This is primarily due to the fact that the direct access scenarios of the original Part 61 analysis methodology do not appear to be sufficiently credible for wastes disposed more than 5 to 10 meters below the ground surface. However, direct access scenarios remain the most reasonable scenarios to use to establish generic concentration limitations. Thus, a credible direct access scenario is needed for waste disposed by deeper methods.

In this scenario, it is assumed that an inadvertant intruder contracts directly with a local contractor to drill a water well to supply the domestic needs of the intruder's family. Unknown to the intruder or the driller, the well is drilled through a near-surface disposal facility and some waste material is brought to the surface.

The following major assumptions are made concerning the drilling scenario.

- The underlying soils are generally fine grained exhibiting low permeabilities and transmissivities.
- The well will be screened in an aquifer underlying the low permeability soils and will be capable of producing approximately 2 gallons per minute (gpm).
- The drill depth will be approximately 200 feet.
- The envisioned well drilling techniques will not be substantially different than those currently used.

E.2.2 Drilling Techniques

The actual drilling technique used will be dependent on local customs for that region. Drillers from different parts of the country seem to prefer different types of equipment.

Selection of the drilling method best suited for a particular job is based on the following factors in approximate order of importance:

(1) Hydrogeologic environment including type of formation, depth of drilling, and depth of screen setting below water table;
(2) Quantity of water required;
(3) Location of drilling, e.g., whether it is accessible by truck; and
(4) Availability of drilling equipment.

The principles of operation, and advantages and disadvantages, of the more common types of drilling techniques suitable for groundwater wells are discussed below. More information can be located in References 1 and 2.

E.2.2.1 Wash Boring

A wash boring is advanced partly by a chopping and twisting action of a chisel-shaped bit and partly by the jetting action of a stream of water pumped through the drill rod and out the bit (see Figure E.1). These operations may be performed entirely by hand, but a small motor-driven winch and pump are generally used.

As the bit penetrates the formations, the washing action of the bit causes the casing to sink. Cuttings are carried to the surface by the water circulating in the annular space between the drill rod and casing. The drill string is lifted and dropped, while the bit rotates, achieving a cutting action and producing a round hole. A closed system is implemented to recirculate the drilling water. Water is pumped from a pit into the drill string and out the bit. After circulating from the bottom to the top of the borehole, the water is conducted back to the pit where the cuttings settle out.

The drill rod is generally a black-iron pipe 1" to 2" in diameter. Casing is required to keep the hole open in soft clays or sand and gravel but is often unnecessary in stiff clays or similar cohesive sediments. If the borehole stays open by itself, the casing and screen are simply lowered and backfilled to construct a well. If casing is required to drill, slip screens are set using the casing pull-back method.

Major advantages are that the method is inexpensive, and that equipment is light and can reach almost any site. Major disadvantages are that it is slow, and cannot penetrate boulders nor wash up coarse gravel.

E.2.2.2 Jet Percussion

Most jet-percussion rigs (see Figure E.2) are moderately-sized units. The tools and drilling action of the jet-percussion method are similar to those described for the wash-boring method. One essential difference, however, is that during drilling the casing is driven with a drive weight and not allowed to advance of its own weight. Normally, this method is used to place 2" diameter casing in shallow, unconsolidated sand formations but has been used to install 3" to 4" diameter casings to depths of 200 feet. The casing pull-back method is recommended for setting screens.

Advantages and disadvantages of the jet percussion technique are similar to those for the wash boring technique: it is inexpensive and the equipment is simple; however, it is slow, and can only be used on unconsolidated sediments or weathered rock.

E.2.2.3 Cable-Tool (Percussion)

In cable-tool or percussion drilling, the hole is deepened by regularly lifting and dropping a heavy string of drilling tools in the borehole (see Figure E.3). The drill bit breaks or crushes hard rock into small fragments and in soft, unconsolidated sediments loosens the material. The up and down action of the drill string mixes the crushed or loosened particles with water to form a slurry. If no water is present in the formation being penetrated, water is added to the borehole. Cuttings are allowed to accumulate until they start to lessen the impact of the bit and are then removed using a bailer or sand pump.

A cable-tool drill string consists of three units: (1) the drill bit, (2) the drill stem, and (3) the rope socket. The bit provides the cutting edge of the drill string, the action of which is increased by the weight of the drill stem. This weight also acts as a stabilizer, keeping the hole straight. The rope socket connects the string of tools to the cable and allows the tools to rotate slightly with respect to the cable.

Drilling jars may be optionally used which are a pair of sliding, linked bars which provide a slack of 6" to 9" in the drill string. These are used primarily in clay or caving formations. If the tools become stuck, the jars permit successive upward blows in the attempt to free them rather than a steady pull on a cable which might part. The shaking and vibrations produced by the jars helps in freeing a stuck drill string.

The bailer consists of a section of pipe with a check valve at the bottom, and is filled by an up-and-down motion in the bottom of the hole. Each time the bailer is dipped, the valve opens, allowing the cuttings slurry to move into it. The up-and-down motion is continued until the bailer is full. At this point, it is brought to the surface and the contents dumped on the ground. A sand pump is a bailer that is fitted with a plunger so that an upward pull on the plunger tends to produce a vacuum that opens the valve and sucks sand or slurried cuttings into the tubing.

When soft caving formations are encountered, it is necessary to drive casing as the hole is advanced to prevent collapse of the hole. Often the drilling can be performed only a few feet below the bottom of the casing. Because the drill bit is lowered through the casing, the hole created by the bit is smaller than the casing. Therefore, the casing (with a sharp, hardened casing shoe on the bottom) must be driven onto the hole. The shoe, in fact, cuts a slightly larger hole than the drill bit.

Major advantages include simple equipment and operation, ability to operate in almost any formation, and small water requirements during drilling. Major disadvantages include slowness, potential difficulty in pulling casing in order to set screen, and the relatively large diameters required (minimum of 4" casing) plus the cost of steel casing (expensive) in comparison with rotary drilling and plastic casings. In addition, cable-tool rigs have largely been replaced by rotary rigs in some parts of the U.S., hence availability may be difficult.

E.2.2.4 Mud Rotary (Hydraulic Rotary)

In this method, shown in Figures E.4 and E.5, a drilling fluid is pumped down the inside of the drill pipe, returning to the surface through the annulus between the drill pipe and the borehole wall. This fluid cools the drill bit, carries the cuttings to the surface, prevents excessive fluid loss into the formation, and prevents the formation from caving. The fluid is then discharged into the mud pit where the cuttings settle out. At the other end of the pit, the fluid is pumped out to circulate down the drill rod again. The drilling fluid may be clear water, water mixed with bentonite, or water mixed with biodegradable organic "mud".

The drill string consists of the bit, a stabilizer, and the drill pipe. Two basic types of bits are used: roller bits in rock, and drag bits in unconsolidated materials. Roller bits have conical rollers with hardened steel teeth of various lengths, spacing, and shape depending upon the type of material to be drilled. Some rollers have inset carbide buttons for drilling in hard rock. As the rollers rotate, they crush and chip the formation material. Drag bits have fixed blades, and the cutting edge is surfaced with carbide or some other abrasion-resistant material.

The bit is attached to a heavy, weighted section of the drill string called a drill collar or stabilizer. This weight, just above the bit, tends to keep the borehole straight and vertical. The drill rod connects this stabilizer to the kelly. The outside diameter of the drill rod ranges from 6 to 11.4 cm. The kelly is a fluted or square bar which passes through a rotary table and imparts a rotary motion to the drill string. When the length of the kelly has been drilled a new section of rod is added and drilling resumed. The rotating drill pipe turns the bit which cuts into the formation and allows the cuttings to be flushed out.

Mud rotary rigs are the most common rig available. Other types of drilling rigs are, however, better suited for certain geologic environments.

Major advantages include its speed, flexibility in final well construction, and the ability to be used in unconsolidated sediments as well as consolidated rocks. Major disadvantages are complex equipment and operation, and higher equipment costs than the previous three methods. Higher equipment costs are compensated, however, by lower labor costs.

E.2.2.5 Other Methods

There are several other methods, such as air rotary, air drilling with casing hammer, and reverse circulation. These are variations of the above methods. They are usually specially manufactured, and expensive. They do not appear to be likely candidates for use by the contractor of an inadvertant intruder.

E.2.3 Summary and Discussion

Primarily based on the fact that it is the most commonly available equipment, mud rotary (or hydraulic rotary) drilling is selected for the drilling scenario. Other assumptions are as follows:

(1) If resistance is encountered during the drilling (as would be the case if the drill bit hit solid rocks or intact metal), the crew would simply move a few yards horizontally and drill a new hole.

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(2) The diameter of the drill hole is assumed to be 8", which would permit the installation of an 4" diameter casing for the well.

(3) The hole is assumed to be drilled to a depth of 200 ft.

(4) Well drilling is only a portion of the well installation process; it usually contains five distinct steps consisting of drilling, casing, installation of the screen, grouting, and developing and pump installation (some of these steps can be performed simultaneously). Drilling takes about 3-4 hours. After the drilling of the hole, the same crew is assumed to complete the well by performing the rest of the above steps. The total time assumed for the activity is 6 hours.

(5) The mud pit dimensions are assumed to be 8' by 9' by 4' deep. This pit has a volume of about 8.2 m^3, and holds drilling fluid up to a level about 1 ft below the top of the pit. The pit is sized so that approximately one-third of the fluid volume in the pit consists of drill cuttings (Refs. 1, 2).

E.3 CORROSION OF METAL

In the original Part 61 analysis methodology, a very simplistic approach was adopted in assessing impacts from the disposal of activated metals. This was primarily due to the fact that they did not constitute a significant portion of the volume of wastes generated. This report considers activated metals from a number of sources in much greater detail. Thus, a better quantification of the radiological impacts from activated metals is needed.

This chapter is divided into two sections. Section E.3.1 is a general review of the corrosion rates of activated metals, primarily carbon and stainless steels. Emphasis is given to the factors that primarily control environmental release. Section E.3.2 considers specific release scenarios including the geometries of individual activated metal waste streams.

E.3.1 Corrosion Rates

Deterioration of carbon and stainless steel components may occur by either atmospheric or aqueous corrosion. Atmospheric corrosion, which is briefly considered when appropriate, is not of primaryinterest in this report. Thus, this section considers those factors influencing corrosion in aqueous environments with the object of estimating probable corrosion rates under a variety of conditions. In the following discussion, carbon steel and stainless steel are considered separately, as these materials differ significantly in corrosion behavior.

Carbon and stainless steels can be affected by "uniform" or "pitting" corrosion. Uniform corrosion means that the surface of the material is uniformly penetrated a certain distance which is usually expressed in mills (1 mill = 0.001 inch) per year (mpy). Pitting corrosion means that certain spots on the surface of the material corrode; it is usually

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expressed as weight loss per year. Some materials, e.g., some stainless steels, are affected only by pitting corrosion; however, uniform corrosion rates are easier to incorporate into a calculational methodology. Thus, in this appendix, pitting corrosion rates will be converted to uniform corrosion rates.

E.3.1.1 Carbon Steel

The largest activated carbon steel component of concern is a reactor pressure vessel. The ensuing discussion will therefore largely focus on this component, although all aspects of corrosion behavior, including corrosion rates, are equally applicable to other carbon steel items.

Reactor vessel walls are typically 6.8 inches thick for a PWR, and 8.5 inches thick for a BWR. Both are clad with an internal lining of type 304 austenitic stainless steel which will be considered separately.

Reactor vessels for light water reactors have been constructed of various carbon and low alloy steels. Depending on the construction specifications, their corrosion rates may vary. For example, early reactor vessels, including those at Dresden, Shippingport, and Indian Point Unit 1, were constructed of carbon or low alloy plate steel, conforming with ASTM Specifications A212 or A302. These steels contain only residual quantities of copper, chromium, and nickel, the elements which most affect atmospheric corrosion resistance of carbon steels.

More recent reactor vessels are constructed of ASTM A533 Grade B, a heat treated low alloy pressure vessel plate steel. The material nominally contains 0.5% nickel as an alloying element. Nickel, in this concentration, produces a measurable improvement in atmospheric corrosion resistance, particularly in marine and industrial atmospheres.

Aqueous corrosion is defined as that resulting from immersion in water. Corrosion of the reactor vessel and other carbon steel components by this mechanism is a possibility. For the purposes of this study, aqueous corrosion will be assumed to take place in fresh water, which is considered to include all nonsaline natural waters, polluted or unpolluted, found in inland bodies. In addition, aqueous corrosion by brackish or salt water will also be considered for comparison purposes.

The corrosion of carbon steel in fresh water is dependent on a number of factors, including mineral content of the water, pH, presence or absence of dissolved oxygen, velocity, and temperature. These factors usually interact such that the corrosion rate of carbon steel in soft fresh water is between 2 and 6 mpy (Ref. 3).

This rate may be slightly lower in hard (high mineral content) water, and will decrease with time in much the same manner as the atmospheric corrosion rate. This behavior is shown in Figure E.6 for carbon steel immersed for periods up to 16 years in Gatun Lake, Panama (Ref.12). Aqueous corrosion of carbon steel typically includes substantial pitting as well as uniform corrosion. Separate curves in Figure E.6 show the changes in pitting and uniform corrosion rates with time. As stated, all corrosion rates based on weight loss (i.e., pitting corrosion) have been converted in this appendix to uniform penetration rates.

The pH value, a measure for expressing hydrogen ion concentration, is one of the most important characteristics of a water affecting its corrosivity. The relationship between pH and corrosion rate is shown in Figure E.7. There is little difference in the corrosion rate of steel in water having a pH between 4.0 and 10.0, a range covering virtually all naturally occurring waters (Ref. 4). It is well established that the corrosion rate of carbon steel increases at pH values lower than 4 and decreases in environments more alkaline than 10. Occasionally, a natural water will be found with a pH more acidic than 4.0. One such example is Monongahela River water, where pH has been measured in the range 3.5 - 4.0, the low pH caused by coal mine drainage into the river. Even in this relatively aggressive water, the corrosion rate after eight years exposure was only 3 mpy (Ref. 5), as shown in Figure E.8.

Although it would be difficult to locate a natural water more acidic than that of the Monongahela River, a scenario could more readily be developed whereby activated carbon steel components are immersed in water more alkaline than pH 10. Large quantities of cement or concrete may be present in the immediate vicinity of these components (e.g., demolition rubble, solidified liquids). Prolonged contact with concrete increases water pH by leaching calcium hydroxide and other alkaline compounds. Should carbon steel components eventually be immersed in water made alkaline by this process, the corrosion rate would be substantially lower than the 2-5 mpy expected for natural fresh water. A quantitative corrosion rate would be difficult to predict as it would be dependent on the pH, or degree of alkalinity, induced by the concrete.

A second major factor influencing corrosion rate in natural waters is the water chemistry, particularly the levels of oxygen and other dissolved gases. At ordinary temperatures in neutral or near neutral water, dissolved oxygen is necessary for appreciable corrosion of steel. In air saturated water, the initial corrosion rate may be about 18 mpy. This rate diminishes over a period of days as an iron oxide (rust) film is formed which acts as a barrier to oxygen diffusion. The steady state corrosion rate decreases to 2 - 5 mpy, tending to be higher with greater relative motion of water with respect to steel (Ref. 4). Since the diffusion rate at steady state is proportional to oxygen concentration, the corrosion rate is also proportional to oxygen concentration. For the purposes of this study, air saturated water, corresponding to approximately 6 ppm dissolved oxygen, may be assumed as this would yield the maximum corrosion rate.

Water hardness, as determined by mineral content, is the other important aspect of water chemistry affecting corrosion rate. Hard waters, that is, those containing high levels of calcium and magnesium salts, tend to deposit a protective layer of calcium carbonate on metal surfaces. The mineral layer impedes oxygen diffusion and thereby reduces corrosion rate. For this reason, the corrosion rate is usually higher in soft waters which do not have the mineral constituents necessary for the formation of this protective scale.

Various methods, based on water chemistry, have been developed to calculate the corrosivity of a given water. Most notable among these is the Langelier Index, which employs pH, hardness, alkalinity, and temperature to calculate an empirical measure of water corrosivity. However, calculations based on water chemistry alone, such as the Langelier Index, do not give actual corrosion rates, which can only be reliably determined from long term exposure tests.

For example, the Langelier Index for water from a river located in the midwest was calculated to be +0.14, a value indicative of a water having neither corrosive nor protective tendencies. Therefore, should carbon steel components ever be immersed in this river water, an average corrosion rate of about 4 mpy could be assumed. This is a conservative estimate, as the influence of concrete would probably raise the pH, thereby reducing the corrosion rate as discussed previously. For similar fresh water sites, even those containing very soft water, it would be difficult to envision a corrosion rate higher than 7 mpy.

Brackish and salt waters differ from fresh water primarily in that they contain up to 3% chloride. The addition of chloride to air saturated fresh water causes an increase in the corrosion rate of carbon steel. The increase is proportional to the amount of chloride added, up to 3% NaCl, the concentration found in sea water (Ref. 4). Laboratory studies have shown that chlorides cause an increase in corrosion rate by interfering with the formation of the barrier rust film which forms on corroding steel. This allows more oxygen to reach the corroding surface and tends to produce a higher corrosion rate.

In spite of wide variations in temperature, salinity, and other factors, long term immersion tests of carbon steel in sea water show a surprising uniformity of rates of attack for specimens exposed at several points throughout the world. A survey of 34 exposure tests shows a corrosion rate spread between 1.0 and 7.7 mpy, with an average of 4.3 mpy (Ref. 3). This study is summarized in Table E-1. It is interesting to note that the 7.7 mpy figure is based on a test of only one year duration, while the lowest rate, 1.0 mpy, is from a 31 year test. It is apparent that the controlling factors interact so as to hold corrosion rate within rather narrow practical limits. As an example, high water temperature, which tends to promote a higher corrosion rate, also serves to encourage the development of protective calcareous deposits and marine growths which stifle attack.

Based on analysis of corrosion data for fresh and salt waters, as discussed above, the time for aqueous corrosion to entirely consume PWR and BWR reactor vessels can be calculated. The results for different assumptions are shown in Table E-2.

The figures in Table E-2 must be viewed with extreme caution. They represent reasonable estimates only for components continually immersed in waters of relatively unchanging composition. They also assume corrosion to be entirely uniform, with no pitting. Actually, corrosion rate of carbon steel is somewhat non-uniform over the corroding surface. Pits may grow at a rate ten times greater than the rate for general corrosion. Hence, a

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component may be perforated, allowing water to enter, well before it is entirely converted to iron oxide. Once a vessel is perforated, corrosion may proceed from both inside and outside, roughly doubling the general corrosion rate. This effect is important in the case of metal equipment with inner cavities, i.e., inner surfaces cannot be neglected.

In addition, unforeseen factors may alter the corrosivity of the water such that present estimates of corrosion rate no longer apply. With these cautions in mind, these figures do provide a general time frame for aqueous corrosion based on water quality considerations.

E.3.1.2 Stainless Steel

Neutron activated stainless steel components include the reactor vessel cladding and internals. The reactor vessel cladding and internals are austenitic stainless steels, type 304 or 316. These alloys are of the 18 chrome, 8 nickel type, with 316 material also containing 2 - 3% molybdenum to provide improved pitting resistance in aggressive environments. For the purposes of this study, only the austenitic stainless steels, types 304 and 316, will be considered as these are the only austenitic stainless steel materials to have been identified in these components.

Austenitic stainless steels corrode in all natural waters exclusively by pitting, with the extent and rate of pitting determined predominantly by the chloride level. For this reason, most corrosion data for stainless steel in aqueous environments is given in terms of pit depth, rather than weight loss which is more indicative of uniform corrosion. Since pitting in stainless steel is a highly localized form of attack, weight loss is invariably low, even in those waters which induce significant pitting. Dispersion of activated corrosion products, however, is a function solely of weight loss, and unrelated to the corrosion mechanism.

Water flow rate can have a significant effect on corrosion of stainless steel. Activated stainless components may be assumed to be immersed in stagnant water, as this represents the condition producing maximum corrosion, and intuitively seems the most likely to occur.

The single factor of most significance to corrosion of austenitic stainless steel in natural water is the chloride level. Long term exposure tests conducted by various investigators show these steels to be unaffected by fresh waters containing up to 50 ppm chloride (Ref. 6). General corrosion rates, based on weight loss after eight years exposure, were calculated to be <0.001 mpy in two tests, one conducted in Rondout Reservoir, New York, and the other in Gatun Lake, Panama (Ref. 7, 12).

These results are in agreement with potentiodynamic polarization data for water containing various chloride levels. Plots of minimum pitting potential versus chloride ion concentration for type 304 steel show the breakthrough potential to be about 0.4 volts when 100 ppm of chloride ion are present (Ref. 8). This potential is considerably higher than that normally found in fresh water, indicating type 304 stainless steel will not pit in waters containing up to at least 100 ppm chloride. Recent electrochemical

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research has related chloride level to the propensity of alloy 304 and 316 to pit, over the temperature range encompassing all naturally occurring waters. It was found that, at room temperature, pitting will not initiate in type 304 until the chloride level exceeds about 150 ppm, while type 316 does not pit at less than 400 ppm chloride (Ref. 9).

These chloride levels are well above those normally found in fresh water. For example, the midwest river water cited above has a chloride level of only 1.5 ppm. All available data from long term exposure tests and accelerated electrochemical tests indicate that austenitic stainless steel will be entirely resistant to corrosion for an indefinitely long time period at this chloride level. Similar behavior would be expected in virtually all other fresh waters.

Water pH has some effect on corrosion of austenitic stainless steels, although minor in comparison with that of the chloride level. In general, a more alkaline environment tends to retard certain forms of corrosion, while low pH enhances corrosion, especially when combined with chlorides. As noted for carbon steel, the probability is extremely remote of exposure to a water with pH so low that it would increase corrosion rate. Highly alkaline water might be encountered, possibly as a result of contact with calcium hydroxide leached from cement. In this case, no corrosion is anticipated.

Weight losses for austenitic stainless steel exposed in seawater are extremely low and attributable entirely to pitting and/or crevice corrosion. These forms of attack are highly localized and may progress at a high rate. Pitting is typically narrow and deep, with pit depths of 0.5 inch having been measured after one year exposure. Pitting data typically show considerable scatter, especially in measurements of pit depth. This is an inherent characteristic of pitting corrosion, related primarily to changes within the pit or crevice rather than the bulk seawater environment. This effect is obscured when weight loss, rather than pitting, data is presented, since weight loss represents the sum of losses from all pits.

Typical weight loss data for type 316 stainless steel, exposed eight years in tropical seawater, is converted to a uniform corrosion rate of only 0.00025 inch/year (Ref. 6). Other investigators report 0.0002 inch/year for type 304 stainless steel (Ref. 10). Corrosion of austenitic stainless steels remains relatively constant at seawater locations throughout the world. It is also unchanged in polluted seawater. The extent of pitting corrosion is influenced by the presence of crevices and marine fouling, which provide sites for the initiation of localized corrosion. However, this effect is not reflected in weight loss data.

Localized corrosion of stainless steel in seawater is significant and measurable. It occurs at a rate sufficient to preclude the use of these materials in many seawater applications. However, when viewed strictly in terms of weight loss (rather than pitting), the uniform corrosion rate is calculated to be exceedingly low, in the range of 0.2 - 0.3 mils/year. Only this figure is of interest in the study, as it represents the rate at which radioactive isotopes are released to the environment. Assuming a corrosion rate of 0.3 mpy, a one inch thick component, for example, would require in excess of 3,300 years to be entirely corroded.

It should be recognized that a prediction of this type is at best a rough estimate. Stainless steel has been in wide use for only forty years. Extrapolation of relatively short term data to a time frame of thousands of years is a speculative undertaking. There is also the possibility that the corrosion rate, calculated from pitting weight loss may not represent a steady state condition. The corrosion rate could increase if pits develop at an accelerating pace after long immersion. There is insufficient data available at present to conclusively eliminate this possibility. With these cautions in mind, a corrosion rate of 0.2 - 0.3 mils/year represents a best guess for types 304 and 316 stainless steel immersed in stagnant seawater.

E.3.1.3 Summary

Projected corrosion rates have been calculated for carbon and stainless steel components exposed to aqueous environments. Consideration has been given to the effects of environmental factors on corrosion rate. This has enabled the corrosion rate to be estimated for a variety of conditions.

Aqueous corrosion of carbon steel is estimated at 4 mpy. This assumes immersion in river water. The corrosion rate does not differ appreciably in other types of water, with virtually all waters, fresh and salt, falling in the range of 1.0 - 7.7 mpy.

Aqueous corrosion of stainless steel differs from that of carbon steel in that corrosion is localized, in the form of pitting. Chloride level in the water is the predominant factor affecting corrosion of stainless steels in natural waters. Virtually all inland fresh waters contain less than 150 ppm chloride, the threshold level necessary to initiate pitting in type 304 stainless steel. Accordingly, stainless steel components immersed in fresh water are not expected to corrode measurably. Seawater and brackish waters are capable of inducing rapid pitting in stainless steels. Since pitting is a highly localized form of attack, general corrosion calculated from weight loss is quite low, estimated to be 0.3 mpy.

E.3.2 Release Scenario Parameters

In the original Part 61 analysis methodology a very simplistic approach was adopted in assessing impacts from the disposal of activated metals. This was primarily due to the fact that they did not constitute a significant portion of the volume of wastes generated. This report considers activated metals from a number of sources in much greater detail. Thus, a better quantification of the radiological impacts from activated metals is needed.

This is accomplished through four waste stream specific values/parameters that are triggered through the accessibility index, I9. This index denotes whether the waste stream is not an activated metal waste stream (an index value of 0) or whether it is (an index value greater than 0). An index value of 1 or higher is used to access the following information concerning the activated metal waste stream:

(1) the corrosion time in years, i.e., the period of time after which the waste stream can be assumed to have corroded completely under fully saturated conditions,
(2) the air dispersibility of the waste stream,
(3) the water solubility of the waste stream, and
(4) the self-shielding afforded against direct radiation by the waste stream.

These parameters are discussed in this section.

E.3.2.1 Corrosion Time

The first property is predicated on the fact that each distinct waste stream, such as a PWR core shroud sectioned in a specific geometry or a BWR reactor vessel sectioned in another specific geometry, will take a different period to corrode entirely. This time period depends on the annual corrosion penetration which in turn depends on the material composition of the waste, and on its geometry.

This property is called the corrosion time, and is denoted by the symbol f_{AC} in years. In this report, it is conservatively assumed that all exposed surfaces of the activated metal waste stream corrode at the assumed corrosion rate. This implies, in most cases, that the waste stream fraction corroded per year is equal to the product of the corrosion rate and the surface to volume ratio of the waste stream. Corrosion time is the inverse of corrosion fraction per year.

Two corrosion rates are used in this report: the carbon steel corrosion rate is assumed to be 4 mills/yr (1.02×10^{-4} meters/yr), and the stainless steel corrosion rate is assumed to be 0.3 mils/yr (7.62×10^{-6} meters/yr).

The second component of this parameter, geometry, is waste stream specific. It is discussed in this section for four waste streams likely to be generated as a result of routine operations and decommissioning of power plants. These waste streams are (1) rods and pipes, e.g., control rods, activated piping, etc., (2) plates, e.g., sectioned reactor vessels, plate control rods, (3) spheres, and (4) stainless steel internals, e.g., activated piping and equipment. These are discussed below.

Rods and Pipes

These wastes can be approximated by either a solid cylinder of specific dimensions or by a cylindrical hollow shell. However, the use of the surface to volume ratio for these wastes does not appear to be reasonable.

For example, if CR denotes the corrosion rate, and assuming a solid rod having a radius of r and a length of H, a corrosion time of r/(2CR) is indicated. However, if it is assumed that corrosion occurs only on the side surface of the rod, this value would be r/CR (twice the previous value). (The diameter of a rod is usually much less than its length.) There is likely to be some corrosion on the ends of the rod; however, this corrosion is not so extensive as to cut the corrosion time in half.

In this report, a corrosion time of r/CR is assumed for solid rods. For pipes having a wall thickness of t, using a similar rationale, a corrosion time of t/(2CR) is assumed.

Plates

These wastes are usually very thin plates for which the use of the surface to volume ratio cannot be justified for reasons similar to those given above. Thus, their thickness, t, is used in conjunction with the corrosion rate, i.e., the corrosion time is conservatively assumed to be t/(2CR).

Spheres

These wastes are small lumps of metal which can be approximated as spheres. The surface to volume ratio for spheres is determined to be 3/r, where r is the radius of the sphere. The corrosion time is then given as V/(SxCR), or r/(3CR).

Stainless Steel Internals

These wastes contain equipment, some piping, some plates, wires, etc. Thus, the geometry of these wastes is much more irregular than for rods, pipes, or plates. In this case, the use of the surface to volume ratio in conjuntion with the corrosion rate is indicated. That is, the corrosion time is assumed to be equal to V/(SxCR).

A study performed for the AIF/NESP on the intact decommissioning concept for power reactors assumed several volume to surface ratios for reactor stainless steel internals. One ratio was for BWRs and the other ratio was for PWRs. These ratios were 1.47 cm and 1.87 cm, respectively (Ref. 12). The arithmetic average of these two values will be used in this report for activated metal wastes (other than rods, pipes, or plates) generated during routine operations and decommissioning. Consequently, the corrosion time for these wastes is assumed to be 2190 years.

E.3.2.2 Dispersibility and Solubility

The second and third properties triggered by the accessibility index I9 depend on the fraction corroded per year, the material properties of the waste, and the chemistry of the corrosion products with regard to the transfer agents. Each distinct waste stream is assigned two accessibility multipliers, f_{AD} and f_{AL}, which are used to multiply the transfer coefficients assumed for routine unsolidified waste streams. The factor f_{AD} replaces the dispersibility multiplier, f_D, for these waste streams, and the factor f_{AL} is used to multiply the M_o fractions used in the groundwater scenarios (see Chapter 4.0).

In this report both f_{AD} and f_{AL} are assumed to be equal to the corrosion fraction per year (inverse of f_{AC}) multiplied by the time at which the scenario occurs up to a maximum value of unity. No special credit is assumed for the dispersibility or the solubility of the corrosion products. This is a conservative assumption since only the noncorroded fraction of the waste stream is exempted from the applicable scenario. However, if data can be found on these properties of the corrosion products, these multipliers can be altered accordingly.

E.3.2.3 Self-Shielding

The fourth property, self-shielding, is a function of the geometry of the waste, its material properties, and its relationship to the exposure point. It is denoted by the symbol f_{AG}, and is calculated for each waste stream and is increased linearly to equal unity when the material is completely corroded ($f_{AG} = 1$ at f_{AC} years).

Shielding and self-shielding have been treated extensively in the literature, especially for the calculation of occupational exposures at reactor power plants. The most useful reference on shielding still appears to be Reactor Shielding Design Manual (frequently referred to as "Rockwell") (Ref. 13). This reference gives extensive formulae, tables and figures on calculation of shielding as well as self-shielding factors for various materials and geometries.

The Rockwell manual was used as the basis for preparation of a computer program, called SACAL, which was created for this study to evaluate the self-shielding factors for four different source geometries. These geometries are: (1) plate, (2) disk, (3) sphere, and (4) cylinder. For each geometry, the following input is required: (1) the geometric dimensions of the waste source (for a plate: width/length/thickness, for a disk: radius/thickness, for a sphere: radius, and for a cylinder: radius/height), (2) the gamma ray energy, and (3) the distance from the source to the exposure point. The code then calculates self-shielding factors for six different material compositions: water, concrete, aluminium, iron, lead, and U-238. A listing of the code can be found in Volume 2, Appendix E.

E.3.2.4 METALS.DAT Parameters

This section summarizes the values assumed for the four I9 parameters considered in this report and stored in the METALS.DAT file. These parameters include the corrosion time (f_{AC}), the dispersibility factor (f_{AD}), the solubility factor (f_{AL}), and the self-shielding factor (f_{AG}).

The eleven activated metal waste streams considered in this report are listed in Table E-3, along with other supplementary information. In this table, the notation "I9" denotes the specific value of the I9 index, while the notation "ISTR" indicates its location number in the data files. The notation "Type" denotes whether the waste stream is assumed to be either stainless steel (S) or carbon steel (C) for purposes of analysis.

The assumed geometries of the individual pieces of metal within the waste packages are also listed. Miscellaneous reactor internals are irregularly shaped pieces of material as discussed in Section E.3.2.1. Fuel hardware (R-FUEHARD, L-FUEHARD) can be visualized as solid lumps of materials and roughly modeled as spheres. The volumes of the spheres are approximated by averaging the hardware masses given in Table A-43 of Appendix A for PWR and BWR fuel assemblies, and then converting the masses to a "full density" volume by dividing by 7.8 g/cm^3. Four decommissioning waste streams can be modeled as plates. Thicknesses for the P-DEACVES and B-DEACVES waste streams are determined using estimates for reactor vessel walls as given in Section E.3.1.1. Thicknesses for core shrouds (P-DECORES and B-DECORES) are obtained from References 14 and 15. Wall thicknesses for reprocessing hulls (R-HULLFRP) are estimated to be roughly 25 mills as obtained from Reference 16.

The corrosion times and self shielding factors are listed in Table E-4 and are estimated using the information and assumptions in Section E.3.2.1, plus the information provided in Table E-3. Corrosion times are based on the assumed material composition (either carbon or stainless steel) plus the assumed dimensions of individual pieces of metal. For example, the corrosion time of the P-DEACVES waste stream is determined to be t/2CR = 17.3/(2 x 0.0102) = 850 years.

The self-shielding factors are more difficult to estimate. A number of individual component pieces may be shipped in the same container, so that there are distinct zones of high and low density materials within the container. The waste geometries are modeled as spheres having the same volumes as the waste containers, and the radionuclide activities within the activated metal components are assumed to be homogenized over these volumes. The average density of the activated metal wastes as smeared over the volumes of the waste containers is assumed to be approximately 1 g/cm^3. This is based on an a very brief review of disposal facility shipment records for 1984.

A sphere having the volume of a 50 ft^3 liner has a radius of 70 cm. Assuming this dimension, an average distance from the source to the exposure point of 100 cm, plus a gamma ray energy of 0.79 Mev, SACAL calculates a self-shielding factor of 0.17. This gamma ray energy corresponds to the average energy per photon emitted for Nb-94, which is the principal long-lived gamma-emitting radionuclide of concern within activated metals. A similar calculation for a sphere having the volume of a 55-gallon drum results in a self-shielding factor of 0.28.

Dispersibility and solubility are assumed to be solely a funtion of the corrosion time as discussed in Section E.3.2.2. The dispersibility and solubility factors are therefore taken to be unity, which merely means that no credit is taken for any potential reduction in dispersibility and solubility provided by the corroded material. Thus, they are not listed in Table E-4.

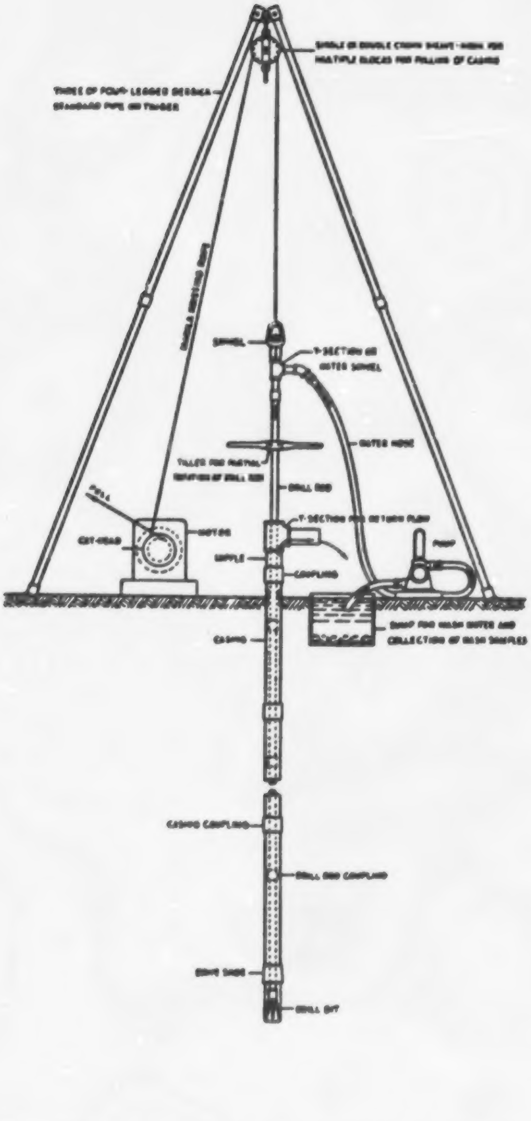
SINGLE OR DOUBLE CROWN SHEAVE - HOOK FOR
MULTIPLE BLOCKS FOR PULLING OF CASING
THREE OR FOUR-LEGGED DERRICK
STANDARD PIPE OR TIMBER
MANILA HOISTING ROPE
SWIVEL
T-SECTION OR
WATER SWIVEL
WATER HOSE
TILLER FOR PARTIAL
ROTATION OF DRILL ROD
DRILL ROD
PULL
CAT-HEAD
MOTOR
T-SECTION FOR RETURN FLOW
NIPPLE
COUPLING
PUMP
SUMP FOR WASH WATER AND
COLLECTION OF WASH SAMPLES
CASING
CASING COUPLING
DRILL ROD COUPLING
DRIVE SHOE
DRILL BIT

FIGURE E.1. Simplified Wash Boring Rig

7d12

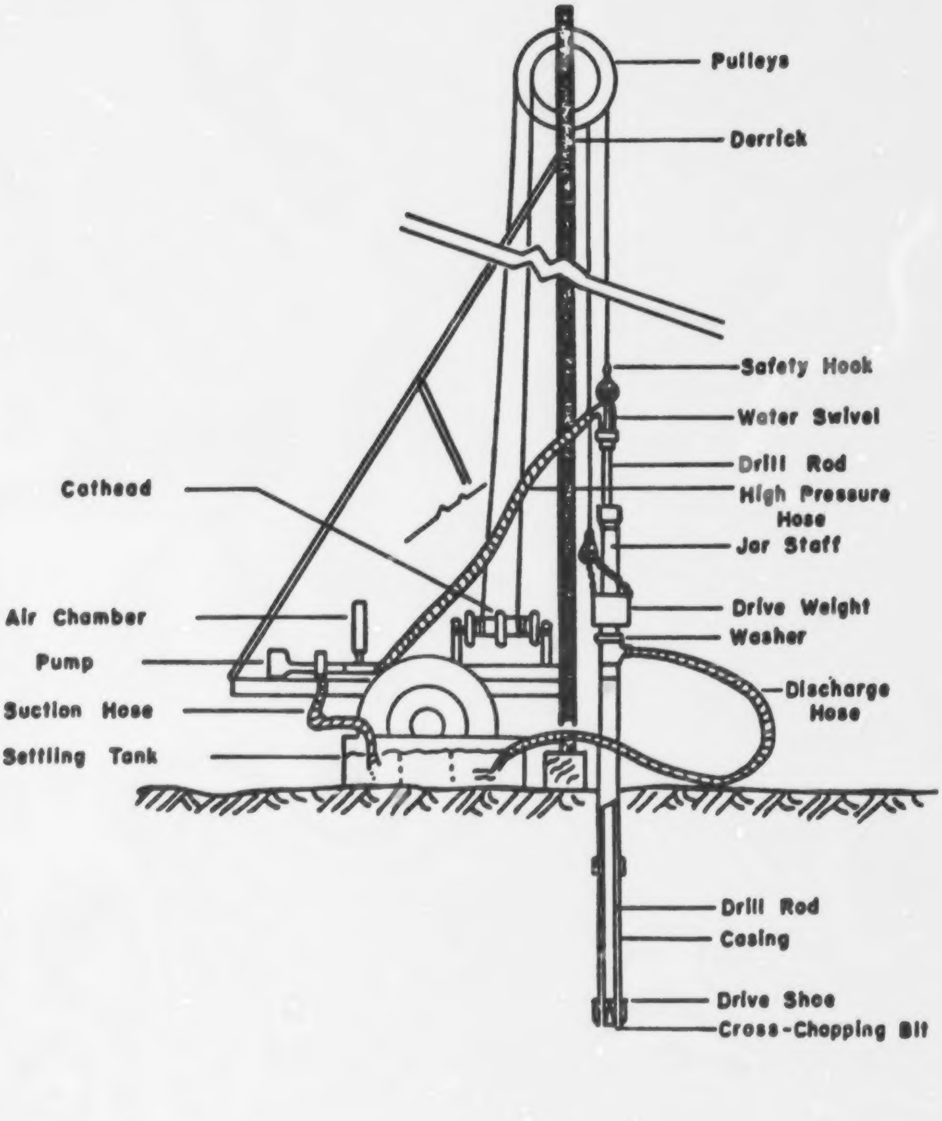
Pulleys
Derrick
Safety Hook
Water Swivel
Drill Rod
High Pressure Hose
Jar Staff
Cathead
Air Chamber
Drive Weight
Washer
Pump
Discharge Hose
Suction Hose
Settling Tank
Drill Rod
Casing
Drive Shoe
Cross-Chopping Bit

FIGURE E.2. Simplified Jet-Percussion

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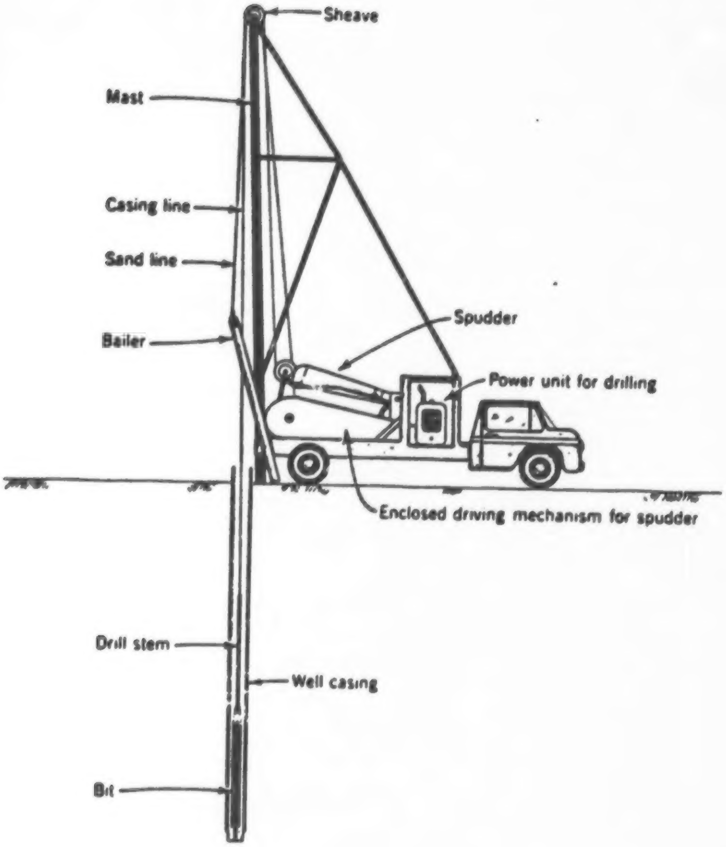
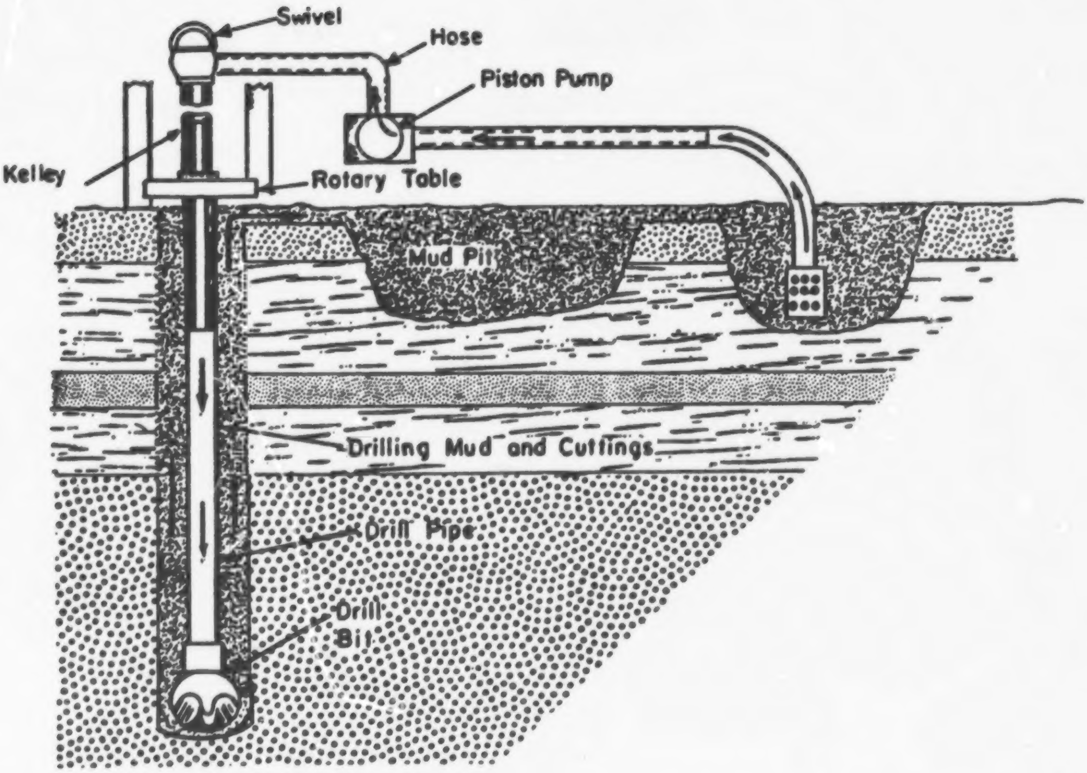

Sheave
Mast
Casing line
Sand line
Bailer
Spudder
Power unit for drilling
Enclosed driving mechanism for spudder
Drill stem
Well casing
Bit

FIGURE E.3. Simplified Cable-Tool Percussion Rig

74.


Swivel
Hose
Piston Pump
Kelley
Rotary Table
Mud Pit
Drilling Mud and Cuttings
Drill Pipe
Drill
Bit

The drilling fluid (or water) is pumped through the swivel and down through the kelley which is turned by the rotary table. The mud then flows down through the drill pipe, out through the bit and back up the hole carrying cuttings which settle out of the mud in the first section(s) of the mud pit.

FIGURE E.4 Mud Rotary Drilling

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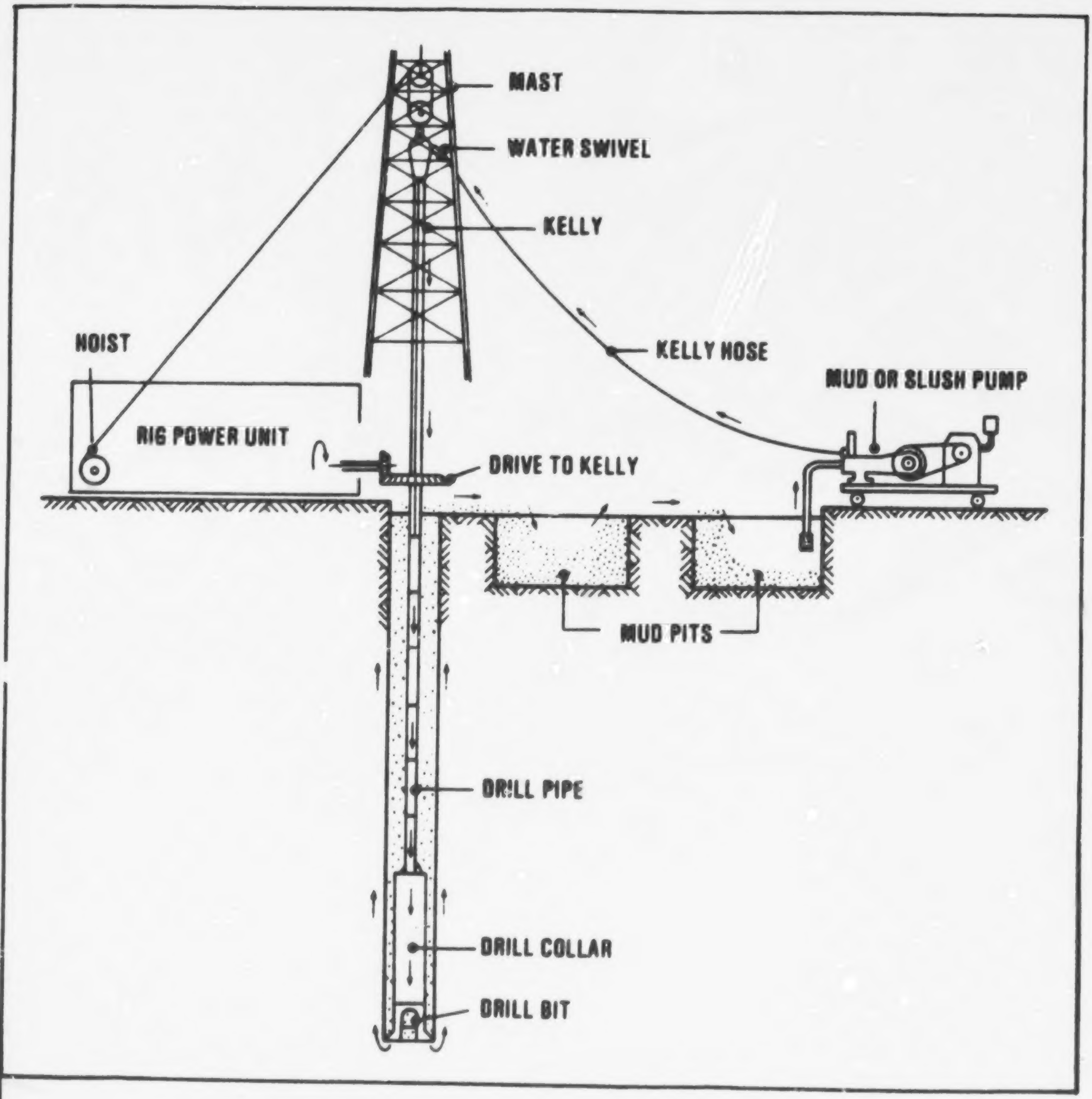
MAST
WATER SWIVEL
KELLY
HOIST
KELLY HOSE
MUD OR SLUSH PUMP
RIG POWER UNIT
DRIVE TO KELLY
MUD PITS
DRILL PIPE
DRILL COLLAR
DRILL BIT

FIGURE E.5. Simplified Hydraulic Rotary

746

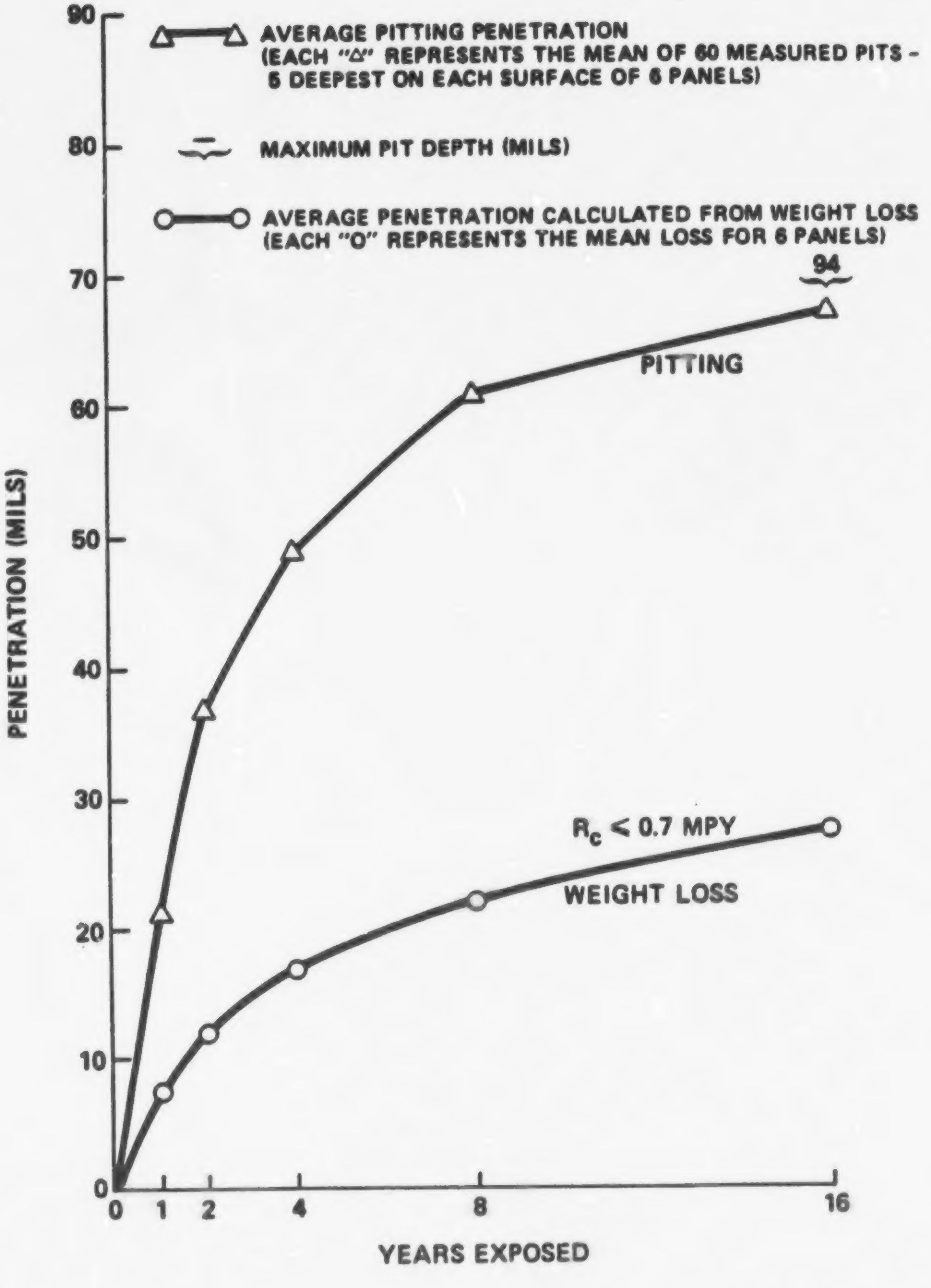

FIGURE E.6. Corrosion of Carbon Steel Continuously Immersed in Fresh Water

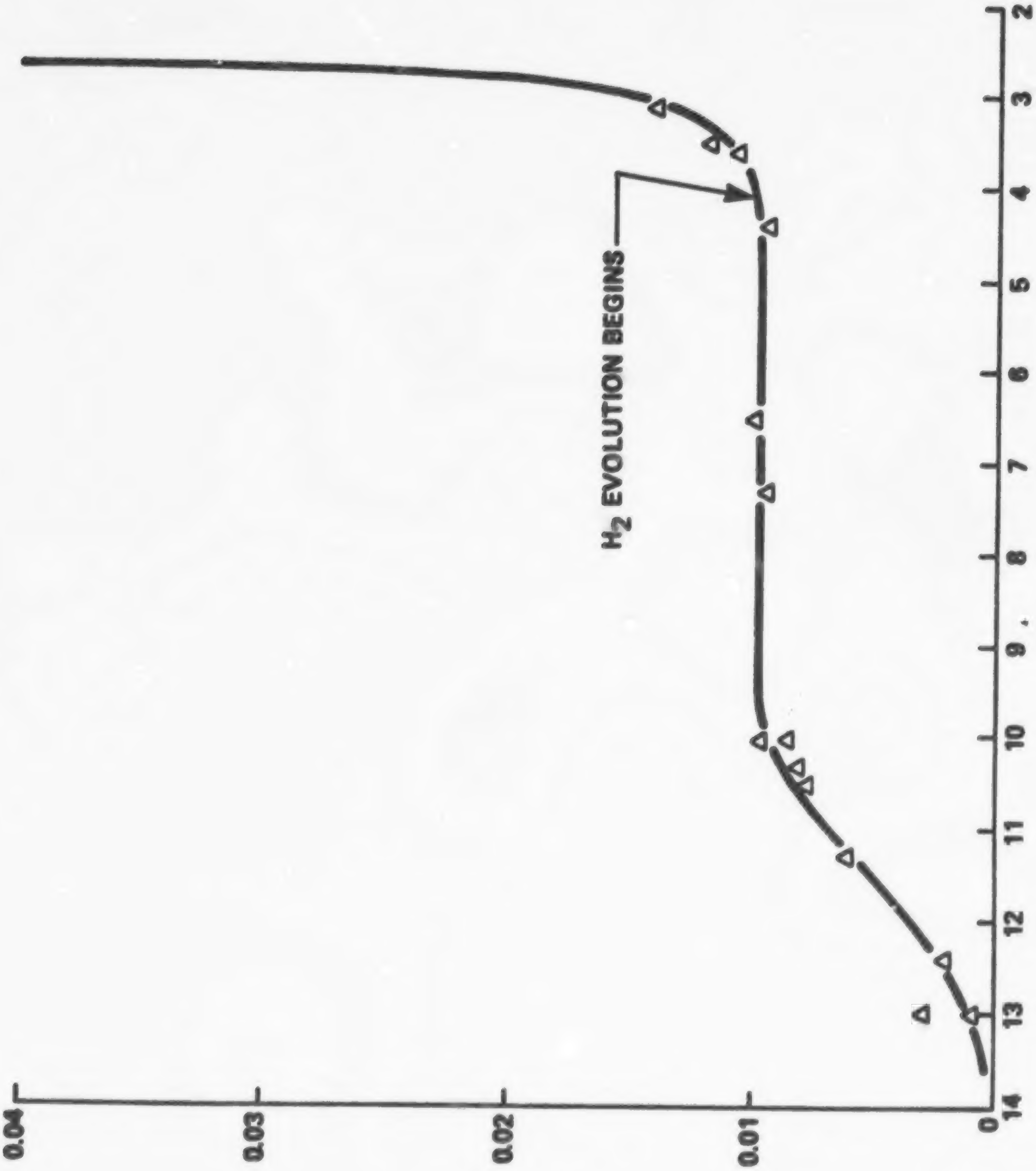

FIGURE E.7. Effects of pH on Corrossion of Iron in Aerated Water, Room Temperature

748

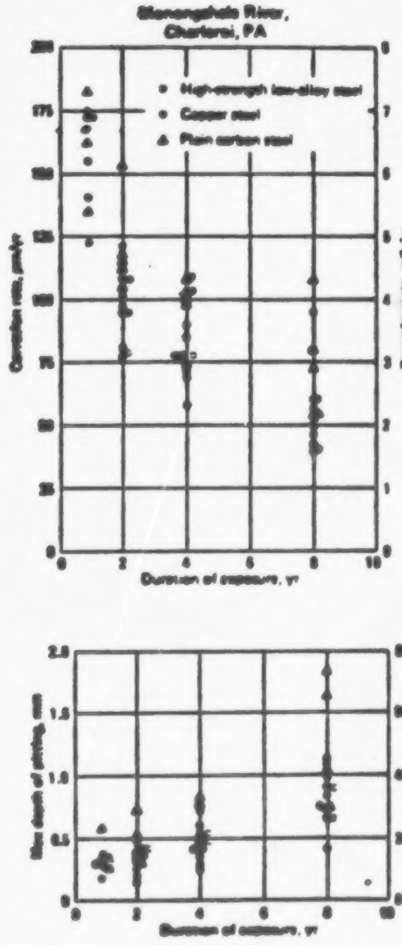

FIGURE E.8. Corrosion in Monongahela River Water

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TABLE E-1 . Corrosion Rate of Carbon Steel at Various Locations(a)

Location	Environment	Corr. Rate, mils/yr (b)
Norman Wells, NWT, Canada	Polar	0.03
Phoenix, AZ	Rural arid	0.18
Esquimalt, Vancouver Island, BC, Canada	Rural marine	0.5
Detroit, MI	Industrial	0.57
Fort Amidor Pier, CZ	Marine	0.57
Morenci, MI	Urban	0.77
Potter Country, PA	Rural	0.8
Westbury, CT	Industrial	0.89
State College, PA	Rural	0.9
Montreal, Que., Canada	Urban	0.9
Durham, NH	Rural	1.1
Middletown, OH	Semi-industrial	1.1
Pittsburgh, PA	Industrial	1.2
Columbus, OH	Industrial	1.3
Trail, BC, Canada	Industrial	1.3
Cleveland, OH	Industrial	1.5
Bethlehem, PA	Industrial	1.5
London, Battersea, England	Industrial	1.8
Monroeville, PA	Semi-industrial	1.9
Newark, NJ	Industrial	2.0
Manila, Phillipine Islands	Tropical marine	2.0
Limon Bay, Panama, CZ	Tropical marine	2.4
Bayonne, NJ	Industrial	3.1
East Chicago, IN	Industrial	3.3
Brazos River, TX	Industrial marine	3.7
Cape Kennedy, FL (c) (60 ft el., 60 yd from ocean)	Marine	5.2
Kure Beach, NC (800 ft from ocean)	Marine	5.8
Cape Kennedy, FL (c) (30 ft el., 60 yd from ocean)	Marine	6.5
Daytona Beach, FL	Marine	11.6
Cape Kennedy, FL (c) (ground level, 60 yd from ocean)	Marine	17.4
Point Reyes, CA	Marine	19.7
Kure Beach, NC (80 ft from ocean)	Marine	21.0
Galeta Point Beach, Panama, CZ	Marine	27.0
Cape Kennedy, FL (c) (beach)	Marine	42.0

(a) From Reference 3.
(b) Two year average.
(c) Now known as Cape Canaveral.

TABLE E-2 . Time for Aqueous Corrosion to Consume Reactor Vessels

Water	Corrosion Rate, inches/yr	Vessel Type	Years For Total Corrosion
Fresh - average	0.004	PWR	1,692
- most corrosive	0.006	PWR	1,128
Salt - average	0.0043	PWR	1,574
- most corrosive	0.0077	PWR	880
Fresh - average	0.004	BWR	2,125
- most corrosive	0.006	BWR	1,416
Salt - average	0.0043	BWR	1,977
- most corrosive	0.0077	BWR	1,104

TABLE E-3 . Summary of Activated Metal Waste Streams Characteristics

I9	Name	ISTR	Packaging	Type	Geometry
1	L-NFRCOMP	12	SL*	S**	Miscellaneous
2	N-HIGHACT	59	DD	C	Miscellaneous
3	R-FUEHARD	81	SL	S	16.5 cm dia. spheres
4	R-HULLFRP	82	SL	S	0.064 cm pipes***
5	P-DECORES	103	SL	S	2.8 cm plates
6	P-DEACINT	104	SL	S	Miscellaneous
7	P-DEACVES	105	SL	C	17.3 cm plates
8	B-DECORES	113	SL	S	5.1 cm plates
9	B-DEACINT	114	SL	S	Miscellaneous
10	B-DEACVES	115	SL	C	21.6 cm plates
11	L-FUEHARD	148	SL	S	16.5 cm dia. spheres

* SL = small (50 ft^3) liner; DD = 55-gallon drum.

** S = stainless steel; C = carbon steel.

*** Wall thickness.

TABLE E-4 . Summary of Activated Metal Waste Streams Corrosion Times and Self-Shielding Factors

I9	Name	Corrosion Time (yrs)	Self-Shielding Factor
1	L-NFRCOMP	2190	0.17
2	N-HIGHACT	164	0.28
3	R-FUEHARD	3610	0.17
4	R-HULLFRP	42	0.17
5	P-DECORES	1840	0.17
6	P-DEACINT	2190	0.17
7	P-DEACVES	850	0.17
8	B-DECORES	3350	0.17
9	B-DEACINT	2190	0.17
10	B-DEACVES	1060	0.17
11	L-FUEHARD	3610	0.17

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13. ABSTRACT *(200 words or less)*

Under contract to the U. S. Nuclear Regulatory Commission, the Envirosphere Company has expanded and updated the impacts analysis methodology used during the development of the 10 CFR Part 61 rule to allow improved consideration of the costs and impacts of treatment and disposal of low-level waste that is close to or exceeds Class C concentrations. These modifications principally include: (1) an update of the low-level radioactive waste source term, (2) consideration of additional alternative disposal technologies, (3) expansion of the methodology used to calculate disposal costs, (4) consideration of an additional exposure pathway involving direct human contact with disposed waste due to a hypothetical drilling scenario, and (5) use of updated health physics analysis procedures (ICRP-30).

Volume 1 of this report describes the calculational algorithms of the updated analysis methodology, while Volume 2 describes the computer codes written to implement the updated analysis methodology plus provides some example problems.

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